Rules and Regulations

Federal Register

Vol. 68, No. 179

Tuesday, September 16, 2003

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50 and 52

RIN 3150–AG76

Combustible Gas Control in Containment

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations for combustible gas control in power reactors applicable to current licensees and is consolidating combustible gas control regulations for future reactor applicants and licensees. The final rule eliminates the requirements for hydrogen recombiners and hydrogen purge systems, and relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their risk significance. This action stems from the NRC's ongoing effort to risk-inform its regulations, and is intended to reduce the regulatory burden on present and future reactor licensees. Additionally, the final rule grants in part and denies in part a petition for rulemaking (PRM–50–68) submitted by Mr. Bob Christie. This notice constitutes final NRC action on PRM–50–68. The final rule also denies part of a petition for rulemaking (PRM–50–71) submitted by the Nuclear Energy Institute. The remaining issue in PRM–50–71 that is not addressed by this final rule will be evaluated in a separate NRC action. The NRC has updated a guidance document, ''Control of Combustible Gas Concentrations in Containment'' to address changes in the rule. A draft regulatory guide containing the revisions was published for comment with the proposed rule.

EFFECTIVE DATE: October 16, 2003.

FOR FURTHER INFORMATION CONTACT: Richard Dudley, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–1116; e-mail: *rfd@nrc.gov.*

SUPPLEMENTARY INFORMATION:

I. Background

On October 27, 1978 (43 FR 50162), the NRC adopted a new rule, 10 CFR 50.44, specifying the standards for combustible gas control systems. The rule required the applicant or licensee to show that during the time period following a postulated loss-of-coolant accident (LOCA), but prior to effective operation of the combustible gas control system, either: (1) An uncontrolled hydrogen-oxygen recombination would not take place in the containment, or (2) the plant could withstand the consequences of an uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions could be shown, the rule required that the containment be provided with an inerted atmosphere to provide protection against hydrogen burning and explosion. The rule defined a release of hydrogen involving up to 5 percent oxidation of the fuel cladding as the amount of hydrogen to be assumed in determining compliance with the rule's provisions. This design-basis hydrogen release was based on the design-basis LOCA postulated by 10 CFR 50.46 and was multiplied by a factor of five for added conservatism to address possible further degradation of emergency core cooling.

The accident at Three Mile Island, Unit 2 involved oxidation of approximately 45 percent of the fuel cladding [NUREG/CR–6197, dated March 1994] with hydrogen generation well in excess of the amounts required to be considered for design purposes by §50.44. Subsequently, the NRC reevaluated the adequacy of the regulations related to hydrogen control to provide greater protection in the event of accidents more severe than design-basis LOCAs. The NRC reassessed the vulnerability of various containment designs to hydrogen burning, which resulted in additional hydrogen control requirements adopted as amendments to §50.44. The 1981 amendment, which added paragraphs (c)(3)(i), (c)(3)(ii), and (c)(3)(iii) to the rule, imposed the following requirements:

(1) An inerted atmosphere for boiling water reactor (BWR) Mark I and Mark II containments,

(2) installation of recombiners for light water reactors that rely on a purge or repressurization system as a primary means of controlling combustible gases following a LOCA, and

(3) installation of high point vents to relieve noncondensible gases from the reactor vessel (46 FR 58484; December 2, 1981).

On January 25, 1985 (50 FR 3498), the NRC published another amendment to § 50.44. This amendment, which added paragraph (c)(3)(iv), required a hydrogen control system justified by a suitable program of experiment and analysis for BWRs with Mark III containments and pressurized water reactors (PWRs) with ice condenser containments. In addition, plants with these containment designs must have systems and components to establish and maintain safe shutdown and containment integrity. These systems must be able to function in an environment after burning and detonation of hydrogen unless it is shown that these events are unlikely to occur. The control system must handle an amount of hydrogen equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region.

When § 50.44 was amended in 1985, the NRC recognized that an improved understanding of the behavior of accidents involving severe core damage was needed. During the 1980s and 1990s, the NRC sponsored a severe accident research program to improve the understanding of core melt phenomena, combustible gas generation, transport and combustion, and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (*e.g.*, NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors.

The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the hydrogen release postulated from a design-basis LOCA was not risk-significant because it was not large enough to lead to early containment failure, and that the risk associated with hydrogen combustion was from beyond design-basis (*e.g.*, severe) accidents. These studies also confirmed the assessment of vulnerabilities that went into the 1981 and 1985 amendments that required additional hydrogen control measures for some containment designs.

II. Rulemaking Initiation

In a June 8, 1999, Staff Requirements Memorandum (SRM) on SECY–98–300, Options for Risk-informed Revisions to 10 CFR Part 50—'' Domestic Licensing of Production and Utilization Facilities,'' the NRC approved proceeding with a study of risk-informing the technical requirements of 10 CFR Part 50. The NRC staff provided its plan and schedule for the study phase of its work to risk-inform the technical requirements of 10 CFR Part 50 in SECY–99–264, ''Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50,'' dated November 8, 1999. The NRC approved proceeding with the plan for risk-informing the Part 50 technical requirements in a February 3, 2000, SRM. Section 50.44 was selected as a test case for piloting the process of risk-informing 10 CFR Part 50 in SECY–00–0086, ''Status Report on Risk-Informing the Technical Requirements of 10 CFR Part 50 (Option 3).''

Mr. Christie of Performance Technology, Inc. submitted letters, dated October 7 and November 9, 1999, that requested changes to the regulations in § 50.44. He requested that the regulations be amended to:

1. Retain the existing requirement in § 50.44(b)(2)(i) for inerting the atmosphere of existing Mark I and Mark II containments.
2. Retain the existing requirement in § 50.44(b)(2)(ii) for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.
3. Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses undertaken pursuant to § 50.44(c).
4. Retain the existing requirements in § 50.44 for high point vents.
5. Eliminate the existing requirement in § 50.44(b)(2) to insure a mixed atmosphere in containment.
6. Eliminate the existing requirement for hydrogen releases during design basis accidents of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.
7. Eliminate the requirement for hydrogen recombiners or purge in LWR containments.
8. Eliminate the existing requirements for hydrogen and oxygen monitoring in LWR containments.
9. Revise GDC 41—Containment Atmosphere Cleanup—to require systems to control fission products and other substances that may be released into the reactor containment for accidents only where there is a high probability that fission products will be released to the reactor containment.

These letters have been treated by the NRC as a petition for rulemaking and assigned Docket No. PRM–50–68. The NRC published a document requesting comment on the petition in the **Federal Register** on January 12, 2000 (65 FR 1829). The issues associated with § 50.44 raised by the petitioner were discussed in SECY–00–0198, ''Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control).'' The final rule and the petition are consistent in many areas, but differ regarding the functional requirements for hydrogen and oxygen monitoring, the requirement for ensuring a mixed atmosphere, the source term of hydrogen for water-cooled reactors to analyze in order to ensure containment integrity, and the need to revise GDC–41. The NRC's detailed basis for including these requirements in the rule is addressed in a subsequent section of this supplementary information.

The NRC also received a petition for rulemaking filed by the Nuclear Energy Institute. The petition was docketed on April 12, 2000, and has been assigned Docket No. PRM–50–71. The petitioner requests that the NRC amend its regulations to allow nuclear power plant licensees to use zirconium-based cladding materials other than zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have received approval by the NRC staff. The petitioner believes the proposed amendment would improve the efficiency of the regulatory process by eliminating the need for individual licensees to obtain exemptions to use advanced cladding materials that have already been approved by the NRC. The change would remove the language in 10 CFR 50.44 regarding the use of zirconium-based cladding materials other than Zircaloy or ZIRLO. The NRC published a document requesting comment on the petition in the **Federal Register** on May 30, 2000 (65 FR 34599). The requested change is unrelated to the risk-informing of 10 CFR 50.44. The NRC addressed the NEI petition in this rulemaking for effective use of resources. Although the final rule does not contain the rule language changes requested by the petitioner, in its revision to 10 CFR 50.44, the NRC eliminated the old language referring to various types of fuel cladding. Thus, the final rule resolves the petitioner's concern regarding § 50.44. The NRC's detailed basis for this decision is addressed in a subsequent section of this supplementary information.

In SECY–00–0198, dated September 14, 2000, the NRC staff proposed a risk-informed voluntary alternative to the current § 50.44. Attachment 2 to that paper, hereafter referred to as the Feasibility Study, used the framework described in Attachment 1 to the paper and risk insights from NUREG–1150 and the IPE programs to evaluate the requirements in § 50.44. The Feasibility Study found that combustible gas generated from design-basis accidents was not risk-significant for any containment type, given intrinsic design capabilities or installed mitigative features. The Feasibility Study also concluded that combustible gas generated from severe accidents was not risk significant for: (1) Mark I and II containments, provided that the required inerted atmosphere was maintained; (2) Mark III and ice condenser containments, provided that the required igniter systems were maintained and operational, and (3) large, dry and sub-atmospheric containments because of the large volumes, high failure pressures, and likelihood of random ignition to help prevent the build-up of detonable hydrogen concentrations.

The Feasibility Study did conclude that the above requirements for combustible gas mitigative features were risk-significant and must be retained. Additionally, the Feasibility Study also indicated that some mitigative features may need to be enhanced beyond current requirements. This concern was identified as Generic Safety Issue-189 (GI–189). The resolution of GI–189 will assess the costs and benefits of improvements to safety which can be achieved by enhancing combustible gas control requirements for Mark III and ice condenser containment designs. The resolution of GI–189 is proceeding independently of this rulemaking. In an SRM dated January 19, 2001, the NRC directed the NRC staff to proceed expeditiously with rulemaking on the risk-informed alternative to § 50.44.

In SECY–01–0162, ''Staff Plans for Proceeding with the Risk-Informed Alternative to the Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors in 10 CFR 50.44,'' dated August 23, 2001, the NRC staff recommended a revised approach to the rulemaking effort. This revised approach recognized that risk-informing Part 50, Option 3 was based on a realistic reevaluation of the basis of a regulation and the application of realistic risk analyses to determine the need for and relative value of regulations that address a design-basis issue. The result of this process necessitates a fundamental reevaluation or ''rebaselining'' of the existing regulation, rather than the development of a voluntary alternative approach to rulemaking. On November 14, 2001, in response to NRC direction in an SRM dated August 2, 2001, the NRC staff published draft rule language on the NRC Web site for stakeholder review and comment. In an SRM dated December 31, 2001, the NRC directed the staff to proceed with the revision to the existing § 50.44 regulations.

III. Final Action

The NRC is retaining existing requirements for ensuring a mixed atmosphere, inerting Mark I and II containments, and hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The NRC is eliminating the design-basis LOCA hydrogen release from § 50.44 and consolidating the requirements for hydrogen and oxygen monitoring into § 50.44 while relaxing safety classifications and licensee commitments to certain design and qualification criteria. The NRC is also relocating and rewording without materially changing the hydrogen control requirements in § 50.34(f) to § 50.44. The high point vent requirements are being relocated from § 50.44 to a new § 50.46a with a change that eliminates a requirement prohibiting venting the reactor coolant system if it could ''aggravate'' the challenge to containment.

Substantive issues are addressed in the following sections.

A. Retention of Inerting, BWR Mark III and PWR Ice Condenser Hydrogen Control Systems, Mixed Atmosphere Requirements, and Associated Analysis Requirements

The final rule retains the existing requirement in § 50.44(c)(3)(i) to inert Mark I and II type containments. Given the relatively small volume and large zirconium inventory, these containments, without inerting, would have a high likelihood of failure from hydrogen combustion due to the potentially large concentration of hydrogen that a severe accident could cause. Retaining the requirement maintains the current level of public protection, as discussed in Section 4.3.2 of the Feasibility Study.

The final rule retains the existing requirements in § 50.44(c)(3)(iv), (v), and (vi) that BWRs with Mark III containments and PWRs with ice condenser containments provide a hydrogen control system justified by a suitable program of experiment and analysis. The amount of hydrogen to be considered is that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). The analyses must demonstrate that the structures, systems and components necessary for safe shutdown and maintaining containment integrity will perform their functions during and after exposure to the conditions created by the burning hydrogen. Environmental conditions caused by local detonations of hydrogen must be included, unless such detonations can be shown unlikely to occur. A significant beyond design-basis accident generating significant amounts of hydrogen (on the order of Three Mile Island, Unit 2, accident or a metal water reaction involving 75 percent of the fuel cladding surrounding the active fuel region) would pose a severe threat to the integrity of these containment types in the absence of the installed igniter systems. Section 4.3.3 of the Feasibility Study concluded that hydrogen combustion is not risk-significant, in terms of the framework document's quantitative guidelines, when igniter systems installed to meet § 50.44(c)(3)(iv), (v), and (vi) are available and operable. The NRC retains these requirements. Previously reviewed and approved licensee analyses to meet the existing regulations constitute compliance with this section. The results of these analyses must continue to be documented in the plant's Updated Final Safety Analysis Report in accordance with § 50.71(e).

The final rule also retains the § 50.44(b)(2) requirement that containments for all currently-licensed nuclear power plants ensure a mixed atmosphere. A mixed containment atmosphere prevents local accumulation of combustible or detonable gases that could threaten containment integrity or equipment operating in a local compartment.

B. Elimination of Design-Basis LOCA Hydrogen Release

The final rule removes the existing definition of a design-basis LOCA hydrogen release and eliminates requirements for hydrogen control systems to mitigate such a release at currently-licensed nuclear power plants. The installation of recombiners and/or vent and purge systems previously required by § 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The NRC finds that this hydrogen release is not risk-significant. This finding is based on the Feasibility Study which found that the design-basis LOCA hydrogen release did not contribute to the conditional probability of a large release up to approximately 24 hours after the onset of core damage. The requirements for combustible gas control that were developed after the Three Mile Island Unit 2 accident were intended to minimize potential additional challenges to containment due to long term residual or radiolytically-generated hydrogen. The NRC found that containment loadings associated with long term hydrogen concentrations are no worse than those considered in the first 24 hours and therefore, are not risk-significant. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by licensee implementation of the severe accident management guidelines (SAMGs) or other ad hoc actions because of the long period of time available to take such action. Therefore, the NRC eliminates the hydrogen release associated with a design-basis LOCA from § 50.44 and the associated requirements that necessitated the need for the hydrogen recombiners and the backup hydrogen vent and purge systems.

In plants with Mark I and II containments, the containment atmosphere is required to be maintained with a low concentration of oxygen, rendering it inert to combustion. Mark I and II containments can be challenged beyond 24 hours by the long-term generation of oxygen through radiolysis. The regulatory analysis for this proposed rulemaking found the cost of maintaining the recombiners exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late time frame. The NRC believes that this conclusion would also be true for the backup hydrogen purge system even though the cost of the hydrogen purge system would be much lower because the system also is needed to inert the containment.

The NRC continues to view severe accident management guidelines as an important part of the severe accident closure process. Severe accident management guidelines are part of a voluntary industry initiative to address accidents beyond the design basis and emergency operating instructions. In November 1994, current nuclear power plant licensees committed to implement severe accident management at their plants by December 31, 1998, using the guidance contained in NEI 91–04, Revision 1, ''Severe Accident Issue Closure Guidelines.'' Generic severe accident management guidelines developed by each nuclear steam system supplier owners group includes either purging and venting or venting the containment to address combustible gas control. On the basis of the industry-wide commitment, the NRC is not requiring such capabilities, but continues to view purging and/or controlled venting of all containment types to be an important combustible gas control strategy that should be considered in a plant's severe accident management guidelines.

C. Oxygen Monitoring Requirements

The final rule amends § 50.44 to codify the existing regulatory practice of monitoring oxygen in currently-licensed nuclear power plant containments that use an inerted atmosphere for combustible gas control. Standard technical specifications and licensee technical specifications currently require oxygen monitoring to verify the inerted condition in containment. Combustible gases produced by beyond design-basis accidents involving both fuel-cladding oxidation and core-concrete interaction would be risk-significant for plants with Mark I and II containments if not for the inerted containment atmosphere. If an inerted containment was to become de-inerted during a significant beyond design-basis accident, then other severe accident management strategies, such as purging and venting, would need to be considered. The oxygen monitoring is needed to implement these severe accident management strategies, in plant emergency operating procedures, and as an input in emergency response decision making.

The final rule reclassifies oxygen monitors as non safety-related components. Currently, as recommended by the NRC's Regulatory Guide (RG) 1.97, oxygen monitors are classified as Category 1. Category 1 is defined as applying to instrumentation designed for monitoring variables that most directly indicate the accomplishment of a safety function for design-basis events. By eliminating the design-basis LOCA hydrogen release, the oxygen monitors are no longer required to mitigate design-basis accidents. The NRC finds that Category 2, defined in RG 1.97, as applying to instrumentation designated for indicating system operating status, to be the more appropriate categorization for the oxygen monitors, because the monitors will still continue to be required to verify the status of the inerted containment. Further, the NRC believes that sufficient reliability of oxygen monitoring, commensurate with its risk-significance, will be achieved by the guidance associated with the Category 2 classification. Because of the various regulatory means, such as orders, that were used to implement post-TMI requirements, this relaxation may require a license amendment at some facilities. Licensees would also need to update their final safety analysis report to reflect the new classification and RG 1.97 categorization of the monitors in accordance with 10 CFR 50.71(e).

D. Hydrogen Monitoring Requirements

The final rule maintains the existing requirement in § 50.44(b)(1) for monitoring hydrogen in the containment atmosphere for all currently-licensed nuclear power plants. Section 50.44(b)(1), standard technical specifications and licensee technical specifications currently contain requirements for monitoring hydrogen, including operability and surveillance requirements for the monitoring systems. Licensees have made commitments to comply with design and qualification criteria for hydrogen monitors specified in NUREG–0737, Item II.F.1, Attachment 6 and in RG 1.97. The hydrogen monitors are required to assess the degree of core damage during a beyond design-basis accident and confirm that random or deliberate ignition has taken place. Hydrogen monitors are also used, in conjunction with oxygen monitors in inerted containments, to guide response to emergency operating procedures. Hydrogen monitors are also used in emergency operating procedures of BWR Mark III facilities. If an explosive mixture that could threaten containment integrity exists, then other severe accident management strategies, such as purging and/or venting, would need to be considered. The hydrogen monitors are needed to implement these severe accident management strategies.

The final rule reclassifies the hydrogen monitors as non safety-related components for currently-licensed nuclear power plants. With the elimination of the design-basis LOCA hydrogen release (see Item B. earlier), the hydrogen monitors are no longer required to support mitigation of design-basis accidents. Therefore, the hydrogen monitors do not meet the definition of a safety-related component as defined in § 50.2. This is consistent with the NRC's determination that oxygen monitors that are used for beyond-design basis accidents need not be safety grade.

Currently, RG 1.97 recommends classifying the hydrogen monitors in Category 1, defined as applying to instrumentation designed for monitoring key variables that most directly indicate the accomplishment of a safety function for design-basis accident events. Because the hydrogen monitors no longer meet the definition of Category 1 in RG 1.97, the NRC believes that licensees' current commitments are unnecessarily burdensome. The NRC believes that Category 3, as defined in RG 1.97, is an appropriate categorization for the hydrogen monitors because the monitors are required to diagnose the course of significant beyond design-basis accidents. Category 3 applies to high-quality, off-the-shelf backup and diagnostic instrumentation. As with the revision to oxygen monitoring, this relaxation may also require a license amendment at some facilities. Licensees will also need to update their final safety analysis report to reflect the new classification and RG 1.97 categorization of the monitors in accordance with 10 CFR 50.71(e).

E. Technical Specifications for Hydrogen and Oxygen Monitors

As discussed in III.C and III.D above, the amended rule requires equipment for monitoring hydrogen in all containments and for monitoring oxygen in containments that use an inerted atmosphere. The rule also requires that this equipment must be functional, reliable, and capable of continuously measuring the concentration of oxygen and/or hydrogen in containment atmosphere following a beyond design-basis accident for combustible gas control and severe accident management, including emergency planning. Because of the importance of these monitors for the management of severe accidents, the NRC staff evaluated whether operability and surveillance requirements for these monitors should be included in the technical specifications.

In order to be retained in the technical specifications, the monitors must meet one of the four criteria set forth by 10 CFR 50.36. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and that functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system or component that operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

As stated in the **Federal Register** notice (60 FR 36953) for the final rule for technical specifications, these criteria were established to address a ''trend toward including in technical specifications not only those requirements derived from the analyses and evaluations included in the safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power plants. This extensive use of technical specifications is due in part to a lack of well-defined criteria (in either the body of the rule or in some other regulatory document) for what should be included in technical specifications.'' As such, the NRC has decided, and established by rule, not to duplicate regulatory requirements in the technical specifications.

Hydrogen and oxygen monitors do not meet criteria 1, 2, or 3 of 10 CFR 50.36 described above. In addition, the Feasibility Study performed by the NRC, and documented in section 4 of Attachment 2 of SECY–00–0198, concluded that the requirement to provide a system to measure the hydrogen concentration in containment does not contribute to the risk estimates for core melt accidents for large dry containments; is not risk significant during the early stages of core melt accidents for Mark I and Mark II containments; and is not risk significant in terms of dealing with the combustion threat of a core melt accident (except for those conditions when the igniters are not operable, *e.g.*, Station Blackout) for Mark III and ice condenser containments. These conclusions were based on the assumptions that Mark I and Mark II containments are inert and hydrogen igniters are operable for Mark III and ice condenser containments. It should be noted that the existing technical specification requirements for hydrogen igniters and for maintaining primary containment oxygen concentration below 4 percent by volume (*i.e.*, inerted), are not being removed; therefore, the conclusions in the Feasibility Study on the risk significance of the hydrogen monitors remain valid. On this basis, the NRC has concluded that hydrogen monitors do not meet criterion 4 of 10 CFR 50.36.

Oxygen monitoring is not the primary means of indicating a significant abnormal degradation of the reactor coolant pressure boundary. Oxygen monitors are used to determine the primary containment oxygen concentration in boiling water reactors. As stated above, the limit for primary containment oxygen concentration for Mark I and II containments will remain in technical specifications; therefore, a technical specification requirement for oxygen monitors would be redundant. In addition, technical specifications for hydrogen igniters for Mark III containments will remain. The oxygen monitors have been shown by probabilistic risk assessment to not be risk-significant. On this basis, the NRC has concluded that oxygen monitors do not meet criterion 4 of 10 CFR 50.36.

The NRC has several precedents regarding not duplicating regulatory requirements for severe accidents in the technical specifications. The Anticipated Transients Without Scram (ATWS) rule, (10 CFR 50.62) requires each pressurized water reactor to have equipment from sensor output to final actuation device, diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment is required to be designed to perform its function in a reliable manner and has no associated requirements incorporated in the technical specifications. The Station Blackout (SBO) rule, (10 CFR 50.63) requires that each light water reactor must be able to withstand and/or recover from a station blackout event. Section 50.63 also states that an alternate ac power source will constitute acceptable capability to withstand station blackout provided an analysis is performed that demonstrates that the plant has this capability from onset of the station blackout until the alternate ac source and required shutdown equipment are started and lined up to operate. Again, no requirements for the alternate ac source are required to be in technical specifications.

NRC experience with implementation of the above regulations for non safety-related equipment has shown that reliability commensurate with severe accident assumptions is assured without including such equipment in technical specifications. According to the ''Final Report—Regulatory Effectiveness of the Station Blackout Rule'' (ADAMS ACCESSION NUMBER: ML003741781), the reliability of the alternate ac power source has improved after implementation of the SBO rule. It states:

''Before the SBO rule was issued, only 11 of 78 plants surveyed had a formal EDG reliability program, 11 of 78 plants had a unit average EDG reliability less that 0.95, and 2 of 78 had a unit average EDG reliability of less that 0.90. Since the SBO rule was issued, all plants have established an EDG reliability program that has improved EDG reliability. A report shows that only 3 of 102 operating plants have a unit average EDG reliability less than 0.95 and above 0.90 considering actual performance on demand, and maintenance (and testing) out of service (MOOS) with the reactor at power.''

Therefore, the NRC staff has concluded that requirements for hydrogen and oxygen monitors can be removed from technical specifications. The basis for this conclusion is:

1. These monitors do not meet the criteria of 10 CFR 50.36,
2. The amended 10 CFR 50.44 requires hydrogen and oxygen monitors to be maintained reliable and functional, and
3. The regulatory precedents set by the treatment of other equipment for severe accidents required by 10 CFR 50.62 and 50.63.

F. Combustible Gas Control Requirements for Future Applicants

Section 50.44(c) of the final rule sets forth combustible gas control requirements for all future water-cooled nuclear power reactor designs with characteristics (*e.g.* type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to currently-licensed light-water reactor designs. The NRC's requirements for future reactors previously specified in § 50.34(f)(2)(ix) have been reworded for conciseness but without material change and relocated to § 50.44(c)(2) to consolidate the combustible gas control requirements in § 50.44 for easier reference. This sub-paragraph requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel clad metal-water reaction and must be capable of precluding uniformly distributed concentrations of hydrogen from exceeding 10 percent (by volume). If these conditions cannot be satisfied, an inerted atmosphere must be provided within the containment. The requirements specified in amended § 50.44(c)(2) are applicable to future water-cooled reactors with the same potential for the production of combustible gas as currently-licensed light-water reactor designs and are consistent with the criteria currently contained in § 50.34(f)(2)(ix) to preclude local concentrations of hydrogen collecting in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate accident mitigating features. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) include the lessening of pressure and temperature loadings on the containment and essential equipment. These requirements reflect the Commission's expectation that future designs will achieve a higher standard of severe accident performance (50 FR 32138; August 8, 1985).

Section 50.44(d) applies to non-water-cooled reactors and water-cooled reactors that have different characteristics regarding the production of combustible gases from current light-water reactors. Because the specific details of the designs and construction materials used in such future reactors cannot now be known, paragraph (d) specifies a general performance-based requirement that future applicants submit information to the NRC indicating how the safety impacts of combustible gases generated during design-basis and significant beyond design-basis accidents are addressed to ensure adequate protection of public health and safety and common defense and security. This information must be based in part upon a design-specific probabilistic risk assessment. The Commission has endorsed the use of PRAs as a tool in regulatory decisionmaking, see *Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement* (60 FR 42622, August 16, 1995), and is currently using PRAs as one element in evaluating proposed changes to licensing bases for currently licensed nuclear power plants, see Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisionmaking: General Guidance* (July 1998) and Standard Review Plan, Chapter 19, ''Use of Probabilistic Risk Assessment in Plant-Specific, Risk Informed Decisionmaking: General Guidance,'' NUREG–0800 (July 1998). The use of PRA methodologies in determining whether severe accidents involving combustible gas must be addressed by future non-water-cooled reactor designs (and water-cooled designs which have different combustible gas generation characteristics as compared with the current fleet of light-water-cooled reactors) is a logical extension of the NRC's efforts to expand the use of PRAs in regulatory decisionmaking.

At this time, the NRC is not able to set forth a detailed description of, or specific criteria for defining a ''significant'' beyond design-basis accident for these future reactor designs, because the fuel and vessel design, cladding material, coolant type, and containment strategy for these reactor designs are unknown at the time of this final rulemaking. Based in part upon the design-specific PRA, the NRC will determine: (i) What type of accident is considered ''significant'' for each future reactor design, (ii) whether combustible gas control measures are necessary, and if so, (iii) whether the combustible gas control measures proposed for each design provide adequate protection to public health and safety and common defense and security. Although it is impossible at this time to provide a detailed description or criteria for determining what constitutes a ''significant'' beyond design-basis accident for the future reactors that are subject to this provision, the NRC nonetheless believes that the concept of ''significant'' with respect to severe accidents has regulatory precedent which will guide the NRC staff's evaluation of the PRA information for future plants. Section 50.34(f)(2)(ix) of the NRC's current regulations already defines what is in essence the significant beyond design-basis accident which future reactor designs comparable to current light-water reactor designs must be capable of addressing, viz., an accident comparable to a degraded core accident at a current light-water reactor in which a metal-water reaction occurs involving 100 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). With respect to other ''beyond design-basis'' accidents, the Commission has addressed anticipated transients without scram (ATWS), and station blackout, which are currently regarded as ''beyond design-basis accidents.'' The nuclear power industry, at the behest of the NRC, has developed severe accident management guidelines to provide for a systematized approach for responding to severe accidents during operations. Finally, the Commission has required all nuclear power plant licensees to implement emergency preparedness planning to address the potential for offsite releases of radiation in excess of 10 CFR Part 100 limits. A careful review of these regulatory efforts discloses a common thread: regulatory actions addressing ''beyond design-basis'' accidents have generally been determined based upon a consideration of probability of the accident, together with consideration of the potential scope and seriousness of the health and property value impacts to the general public. Thus, it is possible to set forth a high-level conceptual description of a ''significant'' beyond design-basis accident involving combustible gas for which the

Commission intends for future non-water-cooled reactor designers to address. First, such an accident would have relatively low probability of occurrence, based upon the PRA, but would not be so small that the accident would be deemed incredible. Second, a ''significant'' beyond design-basis accident involving combustible gas would have serious offsite consequences for the public, involving the potential for death or significant acute or chronic health effects to the general public and/or significant radioactive contamination of offsite property which could result in permanent or long-term commitment of property to nuclear use. Such accidents would typically call for activation of offsite emergency preparedness measures in order to mitigate the adverse effects on public health and safety.

The NRC is currently preparing a Draft Regulatory Guide DG–1122 for public comment, in which the terms, ''significant sequences'' and ''significant contributors'' are expected to be addressed. In addition, as part of the proposed rulemaking for risk-informing 10 CFR § 50.46 the Commission has instructed the NRC staff to develop suitable metrics for determining the appropriate risk cutoff for defining the maximum LOCA size. The metrics are to take into account the uncertainties inherent in development of PRAs. The NRC expects that these regulatory activities will ultimately result in more detailed examples of the ''significant beyond design-basis'' concept to assist a potential applicant in developing the design for a future non-water-cooled reactor (and water-cooled reactor designs which are significantly different in concept from current light-water-cooled reactors), and to guide the NRC's review of an application involving such a design.

G. Clarification and Relocation of High Point Vent Requirements From 10 CFR 50.44 to 10 CFR 50.46a

The final rule removes the current requirements for high point vents from § 50.44 and transfers them to a new § 50.46a. The NRC is relocating these requirements because high point vents are relevant to emergency core cooling system (ECCS) performance during severe accidents, and the final § 50.44 does not address ECCS performance. The requirement to install high point vents was adopted in the 1981 amendment to § 50.44. This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.

The final rule amends the language in § 50.44(c)(3)(iii) by deleting the statement, ''the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident.'' For certain severe accident sequences, the use of reactor coolant system high point vents is intended to reduce the amount of core damage by providing an opportunity to restore reactor core cooling. Although the release of noncondensible and combustible gases from the reactor coolant system will, in the short term, ''aggravate'' the challenge to containment, the use of these vents will positively affect the overall course of the accident. The release of any combustible gases from the reactor coolant system has been considered in the containment design and mitigative features that are required for combustible gas control. Any reactor coolant system venting is highly unlikely to affect containment integrity; however, such venting will reduce the likelihood of further core damage. Because overall plant safety is increased by venting through high point vents, the final rule does not include this statement in § 50.46a.

H. Elimination of Post-Accident Inerting

The final rule no longer provides an option to use post-accident inerting as a means of combustible gas control. Although post-accident inerting systems were permitted as a possible alternative for mitigating combustible gas concerns after the accident at Three Mile Island, Unit 2, no licensee has implemented such a system to date. Concerns with a post-accident inerting system include increase in containment pressure with use, limitations on emergency response personnel access, and cost. Sections 50.44(c)(3)(iv)(D) and 50.34(f)(ix)(D) of the former rule were adopted to address these concerns. On November 14, 2001, draft rule language was made available to elicit comment from interested stakeholders. The draft rule language recommended eliminating the option to use post-accident inerting as a means of combustible gas control and asked stakeholders if there was a need to retain these requirements. Stakeholder feedback supported elimination of the post-accident inerting option and indicated that licensees do not intend to convert existing plants to use post-accident inerting. Because there is no need for the regulations to support an approach that is unlikely to be used, the NRC has decided to eliminate post-accident inerting requirements in the final rule.

IV. Comments and Resolution on Proposed Rule and Draft Regulatory Guide

The 60-day comment period for the proposed rule closed on October 16, 2002. The NRC received 14 letters, from 14 commenters, containing approximately 43 comments on the proposed rule and draft regulatory guide. Seven of the commenters were licensees, two were vendors, two were representatives of utility groups (the Nuclear Energy Institute and the Nuclear Utility Group on Equipment Qualification), two were private citizens, and one was a citizen group, Nuclear Information and Resource Service. All comments were considered in formulating the final rule. Copies of the letters are available for public inspection and copying for a fee at the Commission's Public Document Room, located at 11555 Rockville Pike, Room O–1 F23, Rockville, Maryland 20852.

Documents created or received at the NRC after October 16, 2002, are also available electronically at the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm.html*. From this site, the public can gain entry into the NRC's Agencywide Document Access and Management System (ADAMS), which provides text and image files of NRC's public documents. These same documents also may be viewed and downloaded electronically via the interactive rulemaking Web site established by NRC for this rulemaking at *http://ruleforum.llnl.gov*.

The following sections set forth the resolution of the public comments.

A. General Comments

Many commenters expressed strong support for the rule to improve the regulations in § 50.44 and ''commend[ed] the NRC for developing a rule based on risk-informed and performance-based insights that would eliminate unnecessary regulatory requirements.'' One industry commenter indicated that this rule will enhance public health and safety because it increases the reliability of the hydrogen and oxygen monitoring systems. The Advisory Committee on Reactor

Safeguards (ACRS) stated that the draft proposed rulemaking for risk-informed revisions to 10 CFR 50.44 will provide more effective and efficient regulation to deal with combustible gases in containments.

The NRC also received feedback on several issues for which comments were specifically requested in the draft rule language. The existing rule provides detailed, prescriptive instructions using American Society of Mechanical Engineers (ASME) references for analyzing the performance of boiling water reactor (BWR) Mark III and pressurized water reactor (PWR) ice condenser containments. In the final rule, the NRC has provided an option for a more performance-based approach, which received positive public comment. Based upon stakeholder input, the final rule eliminates the existing references to ASME standards and other prescriptive requirements. The regulatory guide attached to this paper includes the ASME approach as one in which the intent of the regulations could be satisfied.

One private citizen questioned why the NRC was considering relaxing requirements that provide protection against some of the uncertainties and hazards of nuclear power. A citizen group opposed the changes by contending that eliminating the design-basis accident release, relaxing safety classifications, and relaxing licensee commitments to certain design and qualification criteria only benefits the money interests of the licensees. This group also stated its belief that the NRC's reliance on limited Three Mile Island (TMI) data points was insufficient to relax requirements solely to accommodate industry cost cutting strategies.

The NRC is moving to risk-informed, performance-based regulation that takes into account the benefits and consequences of actions by licensees and the NRC. One of the benefits of risk-informed regulation is that it concentrates resources on areas that are more important and minimizes resource allocation on areas that are shown to be less significant. As part of the basis for deciding the level of importance of various areas, during the 1980s and 1990s, the NRC sponsored a severe accident research program to improve the understanding of core melt phenomena, combustible gas generation, transport, and combustion, and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (*e.g.*, NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors. The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the combustible gas release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure, and that the risk associated with gas combustion was from beyond-design-basis (*e.g.*, severe) accidents.

In making its regulatory decisions, the NRC first considers public safety, then other issues such as public confidence and reducing unnecessary regulatory burden. Based upon the results of significant research into design-basis and beyond design-basis accidents, the NRC has determined that a design-basis combustible gas release is not risk-significant and certain beyond design-basis combustible gas releases are risk-significant. Therefore, the NRC is removing the requirements for combustible gas control systems that mitigate consequences of non-risk-significant design-basis accidents which are also not effective in reducing the risk from combustible gas releases in beyond-design-basis accidents.

The citizen group also contended that because GSI–191, ''Assessment of Debris Accumulation on PWR Sump Pump Performance'', is not resolved, removing the hydrogen recombiner requirements and relaxing the hydrogen and oxygen monitoring requirements are premature and constitute a dangerous trend towards risk ''misinformed'' regulation.

The NRC disagrees with the commenter's contention. The NRC's philosophy on all GSIs is to first determine whether the existing situation provides adequate protection of public health and safety, and if there is sufficient margin to allow continued safe operation of the affected plants while seeking a final resolution of the GSI. For GSI–191, the NRC concluded that even though uncertainties remained regarding the debris accumulation issue, adequate protection of public health and safety was maintained. Accordingly, the fact that GSI–191 has not reached final resolution does not present an impediment to the revision to § 50.44.

An industry group requested that the terms ''safety-significant'' and ''industrial'' instead of high and low safety/risk significance be used in this rule and regulatory guide. The NRC disagrees. The terms ''high and low safety/risk significance'' were not included in the proposed rule and are not in the final rule. The term ''safety-significant'', when used in supporting documentation, is used to identify systems, structures, and components (SSCs) that contribute to safety. The term does not confer the level of significance on the SSC. Additionally, the term ''risk significant'' is used to identify those conditions that contribute to risk. Again, no level of significance is assigned by the use of this term. Additionally, the change in terminology requested by the commenter would be inconsistent with the supporting NRC documents and reports. Changing terminology could cause unnecessary confusion on the part of licensees and the public.

B. General Clarifications

One commenter questioned if the draft regulatory guide would become Regulatory Guide 1.7, Revision 3. When the NRC resolves the comments on DG–1117, the guidance will be published as Regulatory Guide 1.7, Revision 3.

A licensee requested that the first sentence of Item 3 of the fourth paragraph of section B of the draft regulatory guide be revised to read: ''The following requirements apply to all construction permits or operating licenses under 10 CFR Part 50, and to all design approvals, design certifications, or combined licenses under 10 CFR Part 52, any of which are issued after the effective date of the rule.'' The NRC agrees that the commenter's request represents a clearer way of expressing the NRC's intent. In addition, the term ''manufacturing licenses'' has been added to make clear that the revised requirements apply to applicants for manufacturing licensees, which was inadvertently omitted from the proposed rule. These changes have been included in both the regulatory guide and in the final rule.

The licensee also requested that the NRC reword the statement in section 5 of the draft regulatory guide to read: ''For future applicants and licensees as defined in Part 50.44(c), the analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.'' Another licensee requested that section C.5, ''Containment Integrity'', should state that it does not apply to currently licensed plants. The NRC disagrees with these requests. Section 5 of DG–1117 was intended to apply to current and future plants. However, the wording was not clear and inadvertently caused some confusion on the applicability of the section. To clarify that section 5 applies to current and future plants, its wording has been revised to more closely reflect the rule intent. This revision removes the following statements from the draft regulatory guide: ''The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.'' The above changes remove the misleading language and clarify the applicability of the section.

C. Monitoring Systems

A private citizen expressed concern about the adequacy and survivability of non safety-related hydrogen and oxygen monitors for assessing hydrogen and oxygen levels after an accident. A reactor licensee stated that the changes to the requirements for hydrogen and oxygen monitoring would actually increase the reliability of hydrogen and oxygen monitoring equipment. A monitor vendor indicated that high-quality commercial grade hydrogen monitors may be susceptible to radiation-induced calibration degradation. The vendor also indicated that these monitors are susceptible to damage from aerosols released during the accident. The vendor believes that commercial grade detectors located inside containment would probably not function in a post-accident environment without verification testing and test-based modifications. The vendor claimed the more severe the accident, the less likely the sensors would properly operate due to increased radiation exposure and increased aerosol loading. In addition, the vendor believes that remote sampling lines for monitors located outside of containment are susceptible to clogging from high-solid aerosols. The vendor suggests it is prudent to retain the safety-related status of hydrogen monitors to ensure comprehensive qualification testing.

The NRC believes that the changes to the requirements for hydrogen and oxygen monitors will continue to ensure acceptable monitor performance. If the changes result in a decrease in monitor reliability, it will not be significant and will not affect public health and safety because the functions served by the monitoring systems are not risk-significant for core melt accident sequences. This conclusion is supported by studies documented in the Feasibility Study (Attachment 2 to SECY–00–0198) which indicate the relatively low risk significance of monitoring systems. Because large, dry and sub-atmospheric containments are robust enough to withstand the effects of hydrogen combustion during full core melt accident sequences, hydrogen monitoring is not risk-significant for these containment designs. For BWR Mark I and Mark II containments, hydrogen monitoring systems are not risk-significant in the early stages of a core melt accident because these containments are inerted. For control of combustible gases generated by radiolysis in the late stage of a core melt accident, oxygen monitors are more important than hydrogen monitors for these designs. For this reason, the design and qualification requirements for oxygen monitors are more stringent than they are for hydrogen monitors. During core melt accidents in BWR Mark III and ice condenser containments, the hydrogen igniter systems are initiated by high containment pressure. Because hydrogen monitors are not needed to initiate or activate any mitigative features during these accidents, they are not risk-significant for reducing the combustible gas threat as long as the hydrogen igniters are operable. If the igniters are not operating (such as during station blackout) hydrogen monitoring does not reduce risk since the containment cannot be purged or vented without electrical power. Nevertheless, the amended rule requires licensees to retain hydrogen monitors (and oxygen monitors in Mark I and Mark II BWRs) for their containments because they are useful in implementing emergency planning and severe accident management mitigative actions for beyond design basis accidents.

As noted in sections III C. and D. of this Supplementary Information, as a consequence of eliminating the design-basis LOCA hydrogen release, the oxygen and hydrogen monitors are no longer required to mitigate potential consequences of combustible gases during design-basis LOCA accidents; thus the monitors are not required to be safety-related and need not meet the procurement, quality assurance, and environmental qualification requirements for safety-related components. Even though amended § 50.44 reclassifies requirements for monitoring systems, the hydrogen and oxygen monitoring systems are still required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment. Thus, licensees must consider the effects of radiation exposure and high-solid aerosols on monitor performance if they will be present in the post-accident environment for the specific type of facility and monitoring system design. The change made by the amended rule is that licensees are no longer required to use only safety-grade monitoring equipment. For a particular facility and monitoring system design, licensees will, in many cases, be able to select appropriate, high quality, commercial-grade monitors that will meet the performance requirements in the rule. In other cases, if no suitable commercial-grade monitors are available, safety-grade monitors may still be necessary. Also, because there are more types and designs of commercial-grade monitors available than there are safety-grade, the ability to use commercial-grade equipment may make it possible for licensees to select a better-suited monitor for their particular application. For example, it is stated in Attachment 2 to SECY–00–0198 that existing safety-grade hydrogen monitors have a limited hydrogen concentration range and are not the optimum choice. Commercial-grade monitors have the ability to monitor a wider range of hydrogen concentration and could be a better solution.

Because the amended rule implements a performance-based requirement for hydrogen and oxygen monitors to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond-design-basis accident environment, licensees will have to ensure that their procurement and quality assurance processes for such equipment address equipment reliability and operability in the beyond design basis accident environmental conditions for the specific facility and monitoring system design. Licensees who do not consider reliability and operability in appropriate environmental conditions when designing and procuring monitoring equipment could be found by NRC inspectors to be in violation of the amended rule.

Another vendor asked if additional requirements beyond commercial grade will be imposed on the monitor's pressure retaining components because the analyzer loop forms part of the containment boundary. The monitor's pressure retaining components must meet current regulations concerning containment penetrations. This vendor also asked if their conclusion that grab samples cannot replace continuous monitoring is correct. The NRC has determined that grab samples cannot replace continuous monitoring. However, grab samples may be taken to verify hydrogen concentrations in the latter stages of the accident response.

A vendor asked if two trains of equipment would be an appropriate solution for ensuring analyzer availability. The NRC cannot respond to such a question without more information about the reliability of each individual train. Licensees are required to meet the requirements of the rule. Individual licensees may determine how they will meet the functionality, reliability, and capability requirements of the rule, using appropriate guidance such as the regulatory guide, and subject to NRC review and inspection.

A licensee requested that section C.2.2 of the draft regulatory guide indicate that oxygen monitors are only required for plants that inerted containments. The NRC agrees with the commenter that oxygen monitors are only required for inerted containments, but disagrees with the suggested addition. The first sentence of section C.2.2 already states: ''The proposed Section 50.44 would require that equipment be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control.'' The final version of the regulatory guide continues to indicate that oxygen monitoring is only necessary for facilities that have inerted containments. Thus, the NRC believes that the existing guidance is sufficient. This licensee also requested that another statement in section C.2.2 of the draft regulatory guide regarding existing oxygen monitoring commitments be clarified to show that these systems meet the intent of the rule. The NRC agrees with the need for clarification. The statement has been revised to read: ''Existing oxygen monitoring systems approved by the NRC prior to the effective date of the rule are sufficient to meet this criterion.''

D. Purge

A licensee stated that the (model) safety evaluation (SE) should address the acceptability of eliminating containment purge as the design basis method for post-LOCA hydrogen control. The NRC disagrees. The NRC model SE only addresses requirements in the standard technical specifications or licensee technical specifications (TS). In this case, the NRC model SE is for the elimination of the requirements of hydrogen recombiners, and hydrogen and oxygen monitors from the TS. Because containment purging requirements are not in the standard technical specifications or licensees' technical specifications, the NRC model SE does not make conclusions regarding the acceptability of eliminating containment purging as the design basis method for post-LOCA hydrogen control. However, the following statement from the Statements of Considerations was added to the model SE to address the comment: ''. . . the NRC eliminated the hydrogen release associated with a design-basis LOCA from § 50.44 and the associated requirements that necessitated the need for the hydrogen recombiners and the backup hydrogen vent and purge systems.''

E. Station Blackout/Generic Safety Issue 189

The citizens group stated that the proposed § 50.44 should require the deliberate ignition systems in Mark III and ice condenser containments to be available during station blackout. This comment pertains to resolution of GSI–189. The NRC disagrees with the commenter. The evaluation and resolution of GSI–189 is ongoing and proceeding independently of the rule as noted in Section II of this Supplementary Information.

F. Containment Structural Uncertainties

The citizens group argues that the NRC does not have an adequate non-destructive tool to eliminate concerns that containments were built with voids in their walls, that all steel reinforcement bar was improperly installed during construction to ensure uniform structural integrity of containment walls, and that the concrete used in containment walls is of sufficient quality that leaching of containment walls has not weakened the structure. The commenter states that without such non-destructive tools, it is unreasonable to reduce the defense-in-depth strategy with the proposed rule. The commenter provided no technical basis or information to support the assertion that containments were inadequately constructed. The commenter also asserts that the proposed rule creates an undue risk to the public health and safety to solely accommodate the financial interest of the regulated industry. Again, no technical basis was provided to support the assertion of increased risk.

The NRC disagrees with the commenter. The NRC relies on several layers of protection to prevent, detect, and repair defects discovered during construction of concrete containments, including voids, improperly installed reinforcement bar, and low quality concrete. These layers of protection include:

(1) The implementation by the licensee of their NRC-approved 10 CFR Part 50, Appendix B, Quality Assurance (QA) program and the licensee's Quality Control (QC) program;

(2) The requirements of 10 CFR 50.55(e) that holders of Construction Permits identify, evaluate, and report defects and failures to comply with NRC requirements associated with substantial safety hazards to the NRC in a timely manner, generally within 60 days; and

(3) The verification by NRC inspectors as defined by the NRC's construction inspection program contained in NRC Inspection Manual Chapter 2512 that the construction is in accordance with approved design documents, that the licensee is properly and effectively implementing their QA/QC program, that construction defects are reported to NRC as required by 10 CFR 50.55(e), and that appropriate corrective actions are taken by the licensee.

Whenever there is a doubt about the proper locations of reinforcing bars, or voids in a concrete containment structure, appropriate non destructive examination methods and conservative analysis are used by the licensees to demonstrate that the containment and its vital components are able to perform their intended functions.

In addition, the pre-operational performance of the Structural Integrity Test (SIT) provides an added assurance by physically demonstrating the overall structural capability of a concrete containment. Also, 10 CFR 50.65, the maintenance rule, requires licensees to monitor the performance or condition of certain structures to provide reasonable assurance that the structures are capable of fulfilling their intended function throughout the life of the plant. Licensees must also periodically inspect and test their containments in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, and Appendix J to 10 CFR Part 50. Finally, at plants that have renewed their licenses, aging management programs are in effect to monitor containment structures to ensure that aging does not significantly degrade their functional capability.

G. PRA/Accident Analysis

An individual submitted questions in three areas. First, the commenter asked why the 30-minute initiation time for initiating hydrogen monitoring was overly burdensome and suggested that the proposed 90-minute initiation time was arbitrary. The NRC disagrees with the commenter. The 30-minute initiation time was developed following the TMI–2 accident based on engineering judgement on the time within which the hydrogen monitors needed to be made functional. Putting this equipment into service within 30 minutes, as directed in NUREG–0737, was found by some utilities during severe accident training (*e.g.*, on nuclear power plant simulators) to be unnecessarily distracting to operators, because it took them away from more important tasks that needed to be implemented in the near term while the monitoring did not need to be initiated for a longer period. The NRC has determined that performance-based functional requirements rather than prescriptive requirements achieve the desired goal of hydrogen monitor functionality while giving licensees an opportunity to better use operators' time during an accident. The noted 90 minutes come from the time licensees found was needed to get the monitors running in a manner that still met the goal of monitoring hydrogen levels and allowed sufficient time for other operator actions based on severe accident emergency operating procedures. Thus, the 90 minute time period was a result of changing to a performance-based approach and was not arbitrarily specified as the time within which the operators had to act.

The individual also stated that the proposed rule was reducing ''defense in depth'' and that if a utility cannot afford to operate and maintain its nuclear power reactors with the requisite caution and oversight, then the utility should not operate them at all. The NRC disagrees with the commenter's assertion that the amended regulations do not provide adequate defense-in-depth. Defense-in-depth continues to be a prime consideration in NRC decision making. The NRC makes its decisions considering public safety first. Only after public safety is ensured are other issues such as public confidence and reduction of unnecessary burden considered. Defense-in-depth is an element of the NRC's safety philosophy that employs successive measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. It provides redundancy as well as the philosophy of a multiple-barrier approach against fission product releases. Defense-in-depth does not mean that equipment installed in a nuclear power plant never should be removed. Adequate defense-in-depth may be achieved through multiple means or paths.

The commenter also questioned whether the NRC staff has adequate data to demonstrate that the amount of residual and radiolytically-generated combustible gases generated during a design-basis LOCA would not be risk-significant—especially if the LOCA occurred in a plant with older fuel and SSCs than were present during the accident at Three Mile Island, Unit 2. The NRC disagrees with the commenter's assertion that insufficient information is known about hydrogen generation to support amending the current regulations. The amount of hydrogen generated during a design-basis LOCA is not affected by the relative age or vintage of reactor fuel or SSCs. The NRC has developed significant data and insights on the behavior of design-basis and severe accidents after the TMI–2 accident. In amending § 50.44 in 1985, the NRC recognized that an improved understanding of the behavior of accidents involving severe core damage was needed. During the 1980s and 1990s, the NRC devoted significant resources and sponsored a severe accident research program to improve the understanding of core melt phenomena; combustible gas generation, transport, and combustion; and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (*e.g.*, NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light water reactors. The result of these studies has been an improved understanding of combustible gas behavior during severe accidents. One of the insights from these studies is confirmation that the hydrogen release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure. In addition, it was found that the vast majority of the risk associated with hydrogen combustion was from beyond design-basis (*e.g.*, severe) accidents. The amended requirements are based on the NRC's careful consideration of the post-Three Mile Island information.

H. Passive Autocatalytic Recombiners

An individual questioned why the United States was allowing the removal of recombiners while the French are requiring the installation of passive autocatalytic recombiners in their reactors. The NRC has determined that passive autocatalytic recombiners (PARs) do not need to be considered for U.S. PWRs with large-dry containments or sub-atmospheric containments. This conclusion was drawn after applying the quantitative and qualitative criteria in the form of a framework for risk-informed changes to technical requirements of 10 CFR Part 50 (*See* attachment 1, SECY–00–0198). The NRC found that hydrogen combustion is not a significant threat to the integrity of large, dry containments or sub-atmospheric containments when compared to the 0.1 conditional large release probability of the framework document. In SECY–00–0198, the NRC also concluded that additional combustible gas control requirements for currently licensed large-dry and sub-atmospheric containments were unwarranted.

I. Reactor Venting

An individual expressed concern for the elimination of the requirement prohibiting venting the reactor coolant system if it would aggravate the challenge to containment. According to the comment, the venting could cause an increase in the radiological effluents released off site and an increase in public exposure. The NRC disagrees with the individual's conclusion. As noted in section III.F of this **SUPPLEMENTARY INFORMATION,** the requirement to install high point vents was imposed by the 1981 amendment to § 50.44. This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.

For certain severe accident sequences, the use of reactor coolant system high point vents is intended to reduce the amount of core damage by providing an opportunity to restore reactor core cooling. Although the release of noncondensible and combustible gases from the reactor coolant system could, in the short term, ''aggravate'' the challenge to containment, the use of these vents will positively affect the overall course of the accident. The release of combustible gases from the reactor coolant system has been considered in the containment design and mitigative features that are required for combustible gas control. Any venting is highly unlikely to affect containment integrity or cause an increase in the radiological effluents released off site that could potentially increase public radiation exposure. However, such venting may reduce the likelihood of further core damage. The reduction in core damage would reduce both the generation of combustible gases and the magnitude of the radiological source term that could be released, thus reducing the potential for public exposure.

An industry organization requested a revision in a statement in section III.F in the statement of considerations (SOC) concerning the purposes of the high point vents from: ''* * * venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged core and thus prevents further accident progression'' to '' * * * the purpose of the high point venting is to ensure that natural circulation cooling is an option for maintaining a long term safe stable state following a core damage accident in which significant amounts of noncondensible gases, such as hydrogen might be generated and retained in the reactor coolant system.'' The NRC disagrees with the comment and believes the current wording is adequate. Other information in section III.F adequately defines the purpose of high point vents by acknowledging their usefulness both for forced circulation scenarios and in the natural circulation mode.

J. Design Basis Accident Hydrogen Source Term

A private citizen questioned that because an unexpected hydrogen bubble and an unexpected hydrogen burn occurred during the accident at Three Mile Island, should hydrogen buildup be considered a known risk for which licensees should try to monitor and control as thoroughly as possible? The NRC agrees with the commenter that hydrogen generation during severe accidents is an expected phenomenon. After the TMI accident, the NRC has sponsored an extensive research program on the behavior of severe accidents. This program was designed improve the understanding of core melt phenomena; combustible gas generation, transport, and combustion; and to develop improved models to predict the progression of severe accidents. The results of this research have been incorporated into various studies (*e.g.*, NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for water-cooled reactors.

The result of these studies has been an improved understanding of combustible gas behavior during severe accidents and confirmation that the combustible gas release postulated from a design-basis LOCA was not risk-significant because it would not lead to early containment failure, and that the risk associated with gas combustion was from beyond-design-basis (*e.g.*, severe) accidents. Thus, the requirements for control and monitoring of combustible gases are being reduced for the non-risk-significant design-basis accident scenarios. The amended regulations are entirely consistent with and justified by the findings of the post-TMI studies.

K. Requested Minor Modifications

An industry group requested that the last paragraph of Section B of the draft regulatory guide be changed to read: ''The treatment requirements for the safety-significant components in the combustible gas control systems, the atmospheric mixing systems and the provisions for measuring and sampling are delineated in Section C, Regulatory Position.'' The NRC disagrees with the requested change. Section 50.44 is being revised to eliminate unnecessary requirements relating to combustible gas control in containment. The remaining requirements have been determined by the NRC to be necessary to mitigate the risk associated with combustible gas generation. The regulatory guide provides recommended treatments for all structures, systems, and components credited for meeting those requirements. Because the regulatory guide is only guidance, licensees are free to devise their own treatments for these structures, systems, and components, subject to NRC review and inspection.

L. Atmosphere Mixing

A private citizen suggested adding criteria to the regulatory guide to assess the adequacy of the performance of atmosphere mixing systems. The NRC disagrees with the commenter that these criteria are needed. The NRC has already evaluated the adequacy of atmosphere mixing at currently operating pressurized and boiling water reactors. However, for future water-cooled reactor designs, the NRC has decided to specify that containments must have the capability for ensuring a mixed atmosphere during ''design-basis and significant beyond design-basis accidents''. Other guidance on determining the adequacy of atmosphere mixing systems is also provided in the rule and the regulatory guide.

An industry group requested that the SOC and regulatory guide be revised to only impose requirements on safety-significant hydrogen (atmospheric) mixing systems. They contend that some large dry containments have hydrogen mixing systems in addition to containment fan cooler units. The fan cooler units are supposedly the prime mode of ensuring a mixed atmosphere; therefore, the hydrogen mixing systems are classified as low safety-significance. The industry group believes that regulatory requirements should not be imposed on low safety-significant equipment. The NRC disagrees with the requested change. Section 50.44 is being revised to eliminate unnecessary requirements relating to combustible gas control in containment. The remaining requirements have been determined by the NRC to be necessary to mitigate the risk associated with combustible gas generation. The regulatory guide provides recommended treatments for all structures, systems, and components credited for meeting those requirements. Because the regulatory guide only provides guidance, licensees are free to devise their own treatments for these structures, systems, and components, subject to NRC review and inspection.

M. Current Versus Future Reactor Facilities

An industry group requested that § 50.44(c) be amended to clarify that its requirements relate only to light-water reactors. The NRC acknowledges that the proposed requirements in § 50.44(c) were largely patterned after light-water reactor requirements and might not be specifically applicable to all types of future light-water and non light-water reactor designs. Therefore, the NRC has modified § 50.44(c) to apply only to future water-cooled reactors with characteristics such that the potential for production of combustible gases during design-basis and significant beyond design-basis accidents is comparable to current light-water reactor designs. In addition, the NRC has added a new paragraph (d) that specifies combustible gas control information to be provided by applicants for future reactor designs when the potential for the production of combustible gases is not comparable to current light-water reactor designs. The purpose of this information is to determine if combustible gas generation is technically relevant to the proposed design; and, if so, to demonstrate that safety impacts of combustible gases generated during design-basis and significant beyond design-basis accidents have been addressed in the design of the facility to ensure adequate protection of public health and safety and common defense and security.

The industry group also commented that the regulatory guide is unclear on what parts are applicable to existing reactors and what parts are applicable to future reactors. The Introduction and section B do not agree. The NRC agrees. The regulatory guide has been modified to clarify the applicability of the revised § 50.44 to present and future water-cooled and non water-cooled reactors. The industry group also noted that the proposed language, the draft regulatory guide, and the proposed change to the Standard Review Plan incorrectly assume that all new reactor designs will be light-water reactors and will present the same combustible gas hazard. Future reactors, whether light-water or non-light-water may use different materials, cooling, or moderating mediums that may not result in the production of the same combustible gases, or quantities of combustible gas as the current light-water reactor designs. The NRC agrees. For the reasons given above, the final rule, the regulatory guide, and the standard review plan have all been modified to clarify their applicability to future reactor designs.

N. Equipment Qualification/Survivability

A licensee suggested adding a clarifying statement to the SOC concerning equipment survivability for Mark III and ice condenser plants. The commenter requested a statement clearly stating that no new equipment survivability requirements are being imposed and that existing equipment survivability and environmental analyses remain valid for compliance with the revised rule. The NRC agrees with commenter that the rule does not impose any additional equipment survivability requirements on licensees; existing equipment survivability and environmental analyses remain valid. The hydrogen and oxygen monitoring systems are required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond design-basis accident environment.

This licensee also noted that, due to the reclassification of the hydrogen and oxygen monitors from RG 1.97 Category I to lower categories, these monitors no longer have to be qualified in accordance with 10 CFR 50.49. The NRC agrees that the monitoring equipment need not be qualified in accordance with § 50.49. The hydrogen and oxygen monitoring systems are still required by the rule to be functional, reliable, and capable of continuously measuring the appropriate parameter in the beyond design-basis accident environment.

The licensee suggested that the NRC clarify that the revised rule will not affect the requirements or environmental conditions used by licensees to demonstrate compliance with § 50.49. The NRC agrees with the commenter that existing licensee analyses and environmental conditions used to establish compliance with 10 CFR 50.49 will not be affected by the amended rule and that no new analyses or environmental conditions are imposed by these amendments to § 50.44.

V. Petitions for Rulemaking—PRM–50–68

The NRC received a petition for rulemaking submitted by Bob Christie of Performance Technology, Knoxville, Tennessee, in the form of two letters dated October 7, 1999, and November 9, 1999. The petition requested that the NRC amend its regulations concerning hydrogen control systems at nuclear power plants. The petitioner believes that the current regulations on hydrogen control systems at some nuclear power plants are detrimental and present a health risk to the public. The petitioner believes that similar detrimental situations may apply to other systems as well (such as the requirement for a 10-second diesel start time). The petitioner believes his proposed amendments would eliminate those situations associated with hydrogen control systems that present adverse conditions at nuclear power plants. The petition was docketed as PRM–50–68 on November 15, 1999. On January 12, 2000 (65 FR 1829), the NRC published a notice of receipt of this petition in the **Federal Register** that summarized the issues it contains.

Specifically, the petitioner performed a detailed review of the San Onofre Task Zero Safety Evaluation Report (Pilot Program for Risk-Informed Performance-Based Regulation) conducted by the NRC staff and dated September 3, 1998, concerning that plant's hydrogen control system. The petitioner requested that the NRC:

1. Retain the existing requirement in § 50.44(b)(2)(i) for inerting the atmosphere of existing Mark I and Mark II containments.
2. Retain the existing requirement in § 50.44(b)(2)(ii) for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.
3. Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses undertaken pursuant to § 50.44(c).
4. Retain the existing requirements in § 50.44 for high point vents.
5. Eliminate the existing requirement in § 50.44(b)(2) for a mixed atmosphere in containment.
6. Eliminate the existing requirement for hydrogen releases during design basis accidents of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.
7. Eliminate the requirement for hydrogen recombiners or purge in LWR containments.
8. Eliminate the existing requirements for hydrogen and oxygen monitoring in LWR containments.
9. Revise GDC 41—Containment Atmosphere Cleanup—to require systems to control fission products and other substances that may be released into the reactor containment for accidents only where there is a high probability that fission products will be released to the reactor containment.
10. Issue an interim policy statement applicable to all NRC staff to ensure that the NRC Executive Director for Operations was promptly notified whenever staff discovered cases where compliance with design-basis accident requirements was detrimental to public health.

The NRC received five comment letters on PRM–50–68. The commenters included two nuclear power plant licensees, a nuclear reactor vendor, a nuclear power plant owners group, and the Nuclear Energy Institute (NEI). Copies of the public comments on PRM–50–68 are available for review in the NRC Public Document Room, 11555 Rockville Pike, Rockville, Maryland. All commenters were supportive of some of the issues raised by the petition. One of the reactor licensees commented that analytical and risk bases exist to support the proposed changes for Mark I Boiling Water Reactor containments. The other licensee endorsed the comments submitted by NEI. The reactor vendor commented that the petitioner's proposal simplifies the language and requirements of the regulation while retaining an equivalent level of safety. However, the vendor also noted that the proposal does not appear to address the structural integrity of the containment as in the existing language at § 50.44(c)(3)(iv). The owner's group commented that the changes requested by the petitioner for large, dry containments were also applicable to ice condenser containments and suggested that the requirement for all hydrogen control measures in § 50.44 be reexamined and made ''consistent with many other portions of plant operation and maintenance.'' The NEI agreed with the petitioner that the San Onofre hydrogen control licensing actions could be applied generically for pressurized water reactors with large, dry (including subatmospheric) containments. One licensee, the reactor vendor and the NEI disagreed with the petitioner's position that an interim policy statement is necessary to instruct the NRC staff how to proceed in instances when ''adherence to design basis requirements would be detrimental to public health.'' The other commenters were silent regarding the request for an interim policy statement.

The NRC has evaluated the technical issues and the associated public comments and has determined that the specific issues contained in PRM–50–68 should be granted in part and denied in part as discussed in the following paragraphs.

Issue 1: Retain the existing requirement for inerting the atmosphere of existing Mark I and Mark II containments.

Resolution of Issue 1: Consistent with the petitioner's request, § 50.44(b)(2)(i) of the final rule retains the current requirement for inerting of existing Mark I and Mark II containments. The NRC's basis for this decision is provided in section III A. of this document.

Issue 2: Retain the existing requirement for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding.

Resolution of Issue 2: Consistent with the petitioner's request, § 50.44(b)(2)(ii) of the final rule retains the above requirement for hydrogen control systems in existing Mark III and PWR ice condenser containments to be capable of handling hydrogen generated by a metal/water reaction involving 75 percent of the fuel cladding. The NRC's basis for this decision is provided in section III A. of this document.

Issue 3: Require all future light water reactors to postulate a 75 percent metal/water reaction (instead of the 100 percent required by the current rule) for analyses under § 50.44(c).

Resolution of Issue 3: The NRC declines to adopt this request. For future water-cooled reactors, the final rule retains the previous requirement to postulate hydrogen generation by a 100 percent metal/water reaction when performing structural analyses of reactor containments under accident conditions. Future containments that cannot structurally withstand the consequences of this amount of hydrogen must be inerted or must be equipped with equipment to reduce the concentration of hydrogen during and following an accident. The NRC's basis for this decision is provided in section III E. of this document.

Issue 4: Retain the existing requirements for high point vents.

Resolution of Issue 4: Consistent with the petitioner's request, the requirements for high point vents in former 10 CFR 50.44(c)(3)(iii) have been retained in the final rule, but have been modified slightly to clarify the acceptable use of these vents during and following an accident. Because the need for high point vents is relevant to ECCS performance during severe accidents and is not pertinent to combustible gas control, these high point venting requirements have been removed from 10 CFR 50.44 and relocated to 10 CFR 50.46a where the remaining requirements for ECCS are located. The basis for this decision is provided in section III F. of this document.

Issue 5 Eliminate the existing requirement in § 50.44(b)(2) to ensure a mixed atmosphere in containment.

Resolution of Issue 5: The NRC declines to adopt this request. The final rule retains the requirement for all containments to ensure a mixed atmosphere to prevent local accumulation of combustible or detonable gasses that could threaten containment integrity or equipment operating in a local compartment. The NRC's basis for retaining this requirement is provided in section III A. of this document.

Issue 6: Eliminate the existing requirement for postulating design basis accident hydrogen releases of an amount equal to that produced by a metal/water reaction of 5 percent of the cladding.

Resolution of Issue 6: The NRC grants this request. The NRC has determined that hydrogen release during design basis accidents is not risk-significant because it does not contribute to the conditional probability of a large release of radionuclides up to approximately 24 hours after the onset of core damage. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by implementation of severe accident management guidelines. The NRC's technical basis for eliminating this requirement is discussed in greater detail in section III B. of this document.

Issue 7: Eliminate the requirement for hydrogen recombiners or purge in light-water reactor containments.

Resolution of Issue 7: The NRC grants this request. As noted in Issue 6 above, the NRC has determined that hydrogen release during design basis accidents is not risk-significant because it does not contribute to the conditional probability of a large release of radionuclides up to approximately 24 hours after the onset of core damage. The NRC believes that accumulation of combustible gases beyond 24 hours can be managed by implementation of severe accident management guidelines. Thus, hydrogen recombiners and hydrogen vent and purge systems are not required. The NRC's basis for eliminating these requirements is discussed in greater detail in section III B. of this document.

Issue 8: Eliminate the existing requirements for hydrogen and oxygen monitoring in light-water reactor containments.

Resolution of Issue 8: The NRC declines to adopt this request. The final rule retains the existing requirement for monitoring hydrogen in the containment atmosphere for all plant designs. Hydrogen monitors are required to assess the degree of core damage during beyond design-basis accidents. Hydrogen monitors are also used in conjunction with oxygen monitors to guide licensees in implementation of severe accident management strategies. Also, the NRC has decided to codify the existing regulatory practice of monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. If an inerted containment became de-inerted during a beyond design-basis accident, other severe accident management strategies, such as purging and venting, would need to be considered. Monitoring of both hydrogen and oxygen is necessary to implement these strategies. The NRC's bases for these requirements are discussed in greater detail in sections III C. and III D. of this document.

Issue 9: Revise GDC 41—Containment Atmosphere Cleanup—to require systems to control fission products and other substances that may be released into the reactor containment for accidents only when there is a high probability that fission products will be released to the reactor containment.

Resolution of Issue 9: The NRC declines to adopt the petitioner's request on this issue. The NRC believes that the amended rule alleviates the need to revise Criterion 41. In a December 4, 2001, letter from the petitioner to the NRC, the petitioner inferred that the intent of the proposed change was to focus Criterion 41 on the containment capability when a severe accident occurs. This concern is addressed in the final § 50.44 that establishes the design criteria for reactor containment and associated equipment for controlling combustible gas released during a postulated severe accident. The General Design Criteria in Appendix A of 10 CFR Part 50 were established to set the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The postulated accidents used in the development of these minimum design criteria are normally design-basis accidents. The NRC believes it is not appropriate to address severe accident design requirements in the General Design Criteria.

Issue 10: The petitioner requested the NRC to issue an interim policy statement applicable to the NRC staff to ensure that the NRC Executive Director for Operations was promptly notified whenever the staff discovered cases where compliance with design-basis accident requirements was detrimental to public health.

Resolution of Issue 10: The petitioner's additional request for an interim policy statement is not part of the petition for rulemaking. Nevertheless, the NRC has evaluated the request and associated public comments and has concluded that hydrogen control requirements referenced by the petitioner have been modified in the final rule so that design basis requirements ensure adequate protection of public health and safety. The NRC also believes that if NRC staff members discover future situations when design basis requirements detract from safety, the staff will elevate these issues for management review; thus, no NRC staff guidance in this area is necessary.

Petition for Rulemaking—PRM–50–71

The NRC also received a petition for rulemaking submitted by NEI. The petition, dated April 12, 2000, was published in the **Federal Register** for public comment on May 31, 2000 (65 FR 34599). The petitioner requested that the NRC amend its regulations to allow nuclear power plant licensees to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided the cladding materials meet the requirements for fuel cladding performance and have been approved by the NRC staff. The petitioner believes the proposed amendment would improve the efficiency of the regulatory process by eliminating the need for individual licensees to obtain exemptions to use advanced cladding materials that have already been approved by the NRC.

Specifically, the petitioner states that the NRC's current regulations require uranium oxide fuel pellets, used in commercial reactor fuel, to be contained in cladding material made of Zircaloy or ZIRLO. The petitioner indicates that the requirement to use either of these materials is stated in § 50.44 and § 50.46. The petitioner notes that subsequent to promulgation of these regulations, commercial nuclear fuel vendors have developed and continue to develop materials other than Zircaloy or ZIRLO that the NRC reviews and approves for use in commercial power reactor fuel. Each of these approvals requires the NRC to grant an exemption to the licensee that requests to use fuel with these cladding materials. The petitioner requests that the NRC amend its regulations to allow licensees discretion to use zirconium-based cladding materials other than Zircaloy or ZIRLO, provided that the cladding materials meet the fuel cladding performance requirements and have been reviewed and approved by the NRC staff. The petitioner notes that during the past nine years there have been at least eight requests for exemptions and that each exemption has cost more than $50,000. The petitioner states that the requests for exemptions have become increasingly more frequent, causing significant administrative confusion and having a potentially adverse effect on efficient and effective use of NRC, licensee, and vendor resources.

The petitioner believes the NRC should amend § 50.44 and § 50.46 to allow the use of other zirconium-based alloys in addition to those specified in the current regulations. The petitioner states that the stated goal of the existing regulations is to ensure adequate cooling for reactor fuel in case of a design-basis accident. However, the petitioner asserts that the proposed amendment does not degrade the ability to meet that goal. The petitioner believes it removes an unwarranted licensing burden without increasing risk to public health and safety.

The NRC received 11 comment letters on PRM 50–71. Seven comments were from nuclear reactor licensees, two from individual members of the public, one from a nuclear reactor vendor and one from a nuclear industry trade association (NEI). Five of the nuclear reactor licensees were supportive of the petition and endorsed the comments and positions provided by NEI in their comments on the petition. One licensee stated that the proposed rule should note that if a fuel vendor's cladding has met the requirements for use on a generic basis, a process for the implementing utility to use that fuel under their existing license already exists. Another licensee agreed that industry needs relief on use of zirconium-based cladding, but because cladding is a critical safety barrier, the basis for relief should come from proven, in-reactor performance. A better approach would be to update the approved list of allowed fuel rod cladding materials as more products demonstrate reliable, in-reactor performance.

Two comments were received from individuals. One individual opposed the petition because it did not contain the specific review and acceptance criteria that NRC would utilize when reviewing and approving future cladding materials under the proposed rule. The commenter also opposed the practice of allowing lead fuel assembly tests to demonstrate performance of new materials in commercial reactors before NRC approval, but also stated that long term performance testing of materials was necessary, must take into account any differences at individual utilities, and must consider future performance in dry cask storage systems. Another individual commented that the petition should be denied because the evaluations of cladding materials do not account for the realities of plant operation under normal conditions and the loss of coolant accident environment. This commenter stated that NRC approval of materials whose properties fell "within" acceptance criteria was unacceptable because an approval might be issued for a material whose properties were "right to the limit" without an adequate margin of safety. With respect to hydrogen generation, the commenter opposed generic approvals of new materials because site-specific material variations might yield unexpected results.

The nuclear reactor vendor supported adoption of the proposed rule changes published in the **Federal Register** and agreed with the suggested revision of § 50.46(e) proposed by NEI in its comments on the document. The vendor also recommended consideration of a direct final rule process to implement the petition. The NEI provided revised wording for proposed language in § 50.46(e) and urged the NRC to promulgate the revision as a direct final rule.

After evaluating the petition and public comments, the NRC has determined that the petition should be denied in part. The final § 50.44 rule has been written so that it does not refer to specific types of zirconium cladding; instead, the rule applies to all boiling and pressurized water reactors. When the NRC approves the use of boiling or pressurized water reactor fuel with other types of cladding, no exemptions from § 50.44 will be needed. Thus, even though the final rule does not contain the language specifically requested to be added by the petitioner, the rule accomplishes the petitioner's intended purpose with respect to § 50.44. Also, the NRC did not utilize the direct final rulemaking process because the other provisions being amended in § 50.44 were too complex to allow the promulgation of a direct final rule. The NRC is making no decision at this time on the part of the petition regarding the request to amend the regulations in § 50.44 to allow the use of other zirconium-based alloys in addition to those specified in the current regulations. The NRC will evaluate that portion of the NEI petition in a separate action.

VII. Section-by-Section Analysis of Substantive Changes

Section 50.34—Contents of Applications; Technical Information

Paragraph (a)(4) on ECCS performance is revised to reference the reactor coolant system high point venting requirements located in § 50.46a. These requirements were relocated to § 50.46a from § 50.44.

Paragraph (g) is redesignated as paragraph (h) and a new paragraph (g) is added, that requires applications for future reactors to include the analyses and descriptions of the equipment and systems required by § 50.44.

Section 50.44—Combustible Gas Control in Containment

Paragraph (a), *Definitions.* Paragraph (a) adds definitions for two previously undefined terms, ''mixed atmosphere,'' and ''inerted atmosphere.''

Paragraph (b), *Requirements for currently-licensed reactors.* This paragraph sets forth the requirements for control of combustible gas in containment for currently-licensed reactors. All BWRs with Mark I and II type containments are required to have an inerted containment atmosphere, and all BWR Mark III type containments and PWRs with ice condenser type containments are required to include a capability for controlling combustible gas generated from a metal water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment integrity. Current requirements in § 50.44(c)(i), (iv), (v), and (vi) are incorporated in to the amended regulation without substantial change. Previously reviewed and installed combustible gas control mitigation features to meet the existing regulations are considered to be sufficient to meet this section. Because these requirements address beyond design-basis combustible gas control, it is acceptable for structures, systems, and components provided to meet these requirements to be non safety-related and may be procured as commercial grade items.

Paragraph (b)(1), *Mixed atmosphere.* The requirement for capability ensuring a mixed atmosphere in all containments is consistent with the current requirement in § 50.44(b)(2) and does not require further analysis or modifications by current licensees. The intent of this requirement is to maintain those plant design features (*e.g.*, availability of active mixing systems or open compartments) that promote atmospheric mixing. The requirement may be met with active or passive systems. Active systems may include a fan, a fan cooler, or containment spray. Passive capability may be demonstrated by evaluating the containment for susceptibility to local hydrogen concentration. These evaluations have been conducted for currently licensed reactors as part of the IPE program.

Paragraph (b)(3) retains the existing requirements for BWR Mark III and PWR ice condenser facilities that do not use inerting to establish and maintain safe shutdown and containment structural integrity to use structures, systems, and components capable of performing their functions during and after exposure to hydrogen combustion.

Paragraph (b)(4)(i) codifies the existing regulatory practice of monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. The rule does not require further analysis or modifications by current licensees but certain design and qualification criteria are relaxed. The rule requires that equipment for monitoring oxygen be functional, reliable and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a beyond design-basis accident. Equipment for monitoring oxygen must perform in the environment anticipated in the severe accident management guidance. The oxygen monitors are expected to be of high-quality and may be procured as commercial grade items. Existing oxygen monitoring commitments for currently licensed plants are sufficient to meet this rule.

Paragraph (b)(4)(ii) retains the requirement in § 50.44(b)(1) for measuring the hydrogen concentration in the containment. The rule does not require further analysis or modifications by current licensees but certain design and qualification criteria are relaxed. The rule requires that equipment for monitoring hydrogen be functional, reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident of comparable severity to the accident at Three Mile Island. Equipment for monitoring hydrogen must perform in the environment anticipated in the severe accident management guidance. The hydrogen monitors may be procured as commercial grade items. Existing hydrogen monitoring commitments for currently licensed plants are sufficient to meet this rule.

Paragraph (b)(5) retains the current analytical requirements in § 50.44(c)(3)(iv) that BWR Mark III and PWR ice condenser containments be provided with a hydrogen control system justified by a suitable program of experiment and analysis that can handle without loss of containment integrity an amount of hydrogen equivalent to that generated by a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel. Existing licensee hydrogen control systems and analyses are expected to be sufficient to demonstrate compliance with this requirement.

Paragraph (c), *Requirements for future water-cooled reactor applicants and licensees.* Paragraph (c) promulgates requirements for combustible gas control in containment for all future water-cooled reactor construction permits or operating licenses under Part 50 and for all water-cooled reactor design approvals, design certifications, combined licenses, or manufacturing licenses under Part 52, whose reactor designs have comparable potential for the production of combustible gases as current light water reactor designs. The current requirements in § 50.34(f)(2)(ix) and (f)(3)(v) are retained without material change, but have been consolidated and reworded to be more concise. Paragraph (c)(1) requires a mixed containment atmosphere during design-basis and significant beyond design-basis accidents. This wording was chosen to specify a mixed atmosphere requirement during important accident scenarios similar to the current requirements for PWR and BWR containments. Paragraph (c)(2) requires all containments to have an inerted atmosphere or limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad coolant reaction, uniformly distributed, to less than 10 percent and maintain containment structural integrity and appropriate accident mitigating features. Structures, systems, and components (SSCs) provided to meet this requirement must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. Equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events (such as station blackout[1] or earthquakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of design-specific reviews which include probabilistic risk-assessment as required by § 52.47(a)(1)(v). Because these requirements address beyond design-basis combustible gas control, SSCs provided to meet these requirements need not be subject to the environmental qualification requirements of § 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; and redundancy/diversity requirements of 10 CFR Part 50, Appendix A. Guidance such as that found in Appendices A and B of RG 1.155, ''Station Blackout,'' is appropriate for equipment used to mitigate the consequences of severe accidents. Paragraph (c) also promulgates requirements for ensuring a mixed atmosphere and monitoring oxygen and hydrogen in containment, consistent with the requirements for current plants set forth in paragraphs (b)(1), and (b)(4)(i) and (ii).

Paragraph (d), *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* A new paragraph (d) is added to specify information that must be submitted by future reactor applicants to determine if combustible gas generation is technically relevant to the proposed design. If combustible gas generation is technically relevant, the applicant must submit additional information to demonstrate that safety impacts of combustible gases generated during design-basis and significant beyond-design-basis accidents have been addressed in the design of the facility to ensure adequate protection of public health and safety and common defense and security. Paragraph (d) is applicable to non water-cooled reactors and water-cooled reactors that have different characteristics regarding the production of combustible gases from current light water reactors. The information must address the potential for producing combustible gases during design basis accidents and significant beyond design-basis accidents comparable to accident scenarios that were evaluated for combustible gas generation at current light water reactors.

Section 50.46a—Acceptance Criteria for Reactor Coolant System Venting Systems

Section 50.46a is a new section that contains the relocated requirements for high point vents currently contained in § 50.44. The amendment includes a change that eliminates a requirement prohibiting venting the reactor coolant system if it could ''aggravate'' the challenge to containment. Any venting is highly unlikely to affect containment integrity; however, such venting will reduce the likelihood of further core damage. The NRC continues to view use of the high point vents as an important strategy that should be considered in a plant's severe accident management guidelines.

Section 52.47—Contents of Applications

Section 52.47 is amended to eliminate the reference to paragraphs within § 50.34(f) for technically relevant requirements for combustible gas control in containment for future design certifications. Under the final rule, the technical requirements for combustible gas control will be set forth in § 50.44, rather than in § 50.34(f).

VIII. Availability of Documents

The NRC is making the documents identified below available to interested persons through one or more of the following methods as indicated.

Public Document Room (PDR). The NRC Public Document Room is located at One White Flint North, Public File Area O 1F21, 11555 Rockville Pike, Rockville, Maryland.

Rulemaking Web site (Web). The NRC's interactive rulemaking Web site is located at *http://ruleforum.llnl.gov.* These documents may be viewed and downloaded electronically via this Web site.

NRC's Electronic Reading Room (ERR). The NRC's public electronic reading room is located at *http://www.nrc.gov/NRC/ADAMS/index.html.* (Provide accession number for each document.)

The NRC staff contact (NRC Staff). Richard Dudley, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; telephone (301) 415–1116; e-mail *rfd@nrc.gov.*

Document	PDR	Web	ERR	NRC staff
Comments received	X	X	X	....
Regulatory Analysis	X	X	ML031640482	....
RG 1.7, Rev. 3	X	X	ML031640498	X
Rev. SRP, Section 6.2.5	X	X	ML031640518	X

A free single copy of Regulatory Guide 1.7 may be obtained by writing to the Office of the Chief Information Officer, Reproduction and Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or E-mail: *DISTRIBUTION@nrc.gov* or Facsimile: (301) 415–2289.

Copies of NUREGS may be purchased from The Superintendent of Documents, U.S. Government Printing Office, Mail Stop SSOP, Washington, DC 20402–0001; Internet: bookstore@gpo.gov; (202) 512–1800. Copies are also available from the National Technical Information Service, Springfield, VA 22161–0002; *http://www.ntis.gov;* 1–800–533–6847 or, locally, (703) 605–6000. Some publications in the NUREG series are posted at NRC's technical document Web site *http://www.nrc.gov/NRC/NUREGS/indexnum.html.*

IX. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104–113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless using such a standard is inconsistent with applicable law or is otherwise impractical. In this final rule, the NRC is using the following Government-unique standard: 10 CFR 50.44, U.S.

[1] Section 50.44 does not require the deliberate ignition systems used by BWRs with Mark III type containments and PWRs with ice condenser type containments to be available during station blackout events. The deliberate ignition systems should be available upon the restoration of power. Additional guidance concerning the availability of deliberate ignition systems during station blackout sequences is being developed as part of the NRC review of Generic Safety Issue 189: ''Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident.''

Nuclear Regulatory Commission, October 27, 1978 (43 FR 50163), as amended. No voluntary consensus standard has been identified that could be used instead of the Government-unique standard.

X. Finding of No Significant Environmental Impact: Environmental Assessment

The NRC has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The basis for this determination reads as follows:

This action endorses existing requirements and establishes regulations that reduce regulatory burdens for current and future licensees and consolidates combustible gas control regulations for future reactor applicants and licensees. This action stems from the NRC's ongoing effort to risk-inform its regulations. The final rule reduces the regulatory burdens on present and future power reactor licensees by eliminating the LOCA design-basis accident as a combustible gas control concern. This change eliminates the requirements for hydrogen recombiners and hydrogen purge systems and relaxes the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with their safety and risk significance.

This action does not significantly increase the probability or consequences of an accident. No changes are being made in the types or quantities of radiological effluents that may be released off site, and there is no significant increase in public radiation exposure because there is no change to facility operations that could create a new or affect a previously analyzed accident or release path. There may be a reduction of occupational radiation exposure since personnel will no longer be required to maintain or operate, if necessary, the hydrogen recombiner systems which are located in or near radiologically controlled areas.

With regard to non-radiological impacts, no changes are being made to non-radiological plant effluents and there are no changes in activities that would adversely affect the environment. Therefore, there are no significant non-radiological impacts associated with the proposed action.

The primary alternative to this action would be the no action alternative. The no action alternative would continue to impose unwarranted regulatory burdens for which there would be little or no safety, risk, or environmental benefit.

The determination of this environmental assessment is that there is no significant offsite impact to the public from this action.

The NRC requested the views of the States on the environmental assessment for this rule. No comments were received.

XI. Paperwork Reduction Act Statement

This final rule decreases the burden on new applicants to complete the hydrogen control analysis required to be submitted in a license application, as required by sections 50.34 or 52.47. The public burden reduction for this information collection is estimated to average 720 hours per request. Because the burden for this information collection is insignificant, Office of Management and Budget (OMB) clearance is not required. Existing requirements were approved by the Office of Management and Budget, approval numbers 3150–0011 and 3150–0151.

XII. Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis

The NRC has prepared a regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the NRC. The regulatory analysis is available as indicated under the Availability of Documents heading of the Supplementary Information section.

XIV. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XV. Backfit Analysis

The NRC has determined that the backfit rule does not apply to this final rule; and therefore, a backfit analysis is not required for this final rule because these amendments do not impose more stringent safety requirements on 10 CFR Part 50 licensees. For current licensees, the amendments either maintain without substantive change existing requirements or provide voluntary relaxations to current regulatory requirements. Voluntary relaxations (*i.e.*, relaxations that are not mandatory) are not considered backfitting as defined in 10 CFR 50.109(a)(1). For future applicants and future licensees, the amendments also do not involve backfitting as defined in 10 CFR 50.109(a)(1) because the changes have only a prospective effect on future design approval and design certification applicants and future applicants for licensees under 10 CFR Part 50 and 52. As the Commission has indicated in other rulemakings, sec., *e.g.*, 54 FR 15372, April 18, 1989 (Final Part 52 Rule), the expectations of future applicants are not protected by the Backfit Rule. Therefore, the NRC has not prepared a backfit analysis for this final rule.

XVI. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire protection, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and record keeping requirements.

10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and record keeping requirements, Standard design, Standard design certification.

■ For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553, the

NRC is adopting the following amendments to 10 CFR Parts 50 and 52.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

■ 1. The authority citation for Part 50 continues to read as follows:

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95–601, sec. 10, 92 Stat. 2951, as amended by Pub. L. 102–486, sec. 2902, 106 Stat. 3123 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under Pub. L. 97–415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80—50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

■ 2. In § 50.34, paragraph (a)(4) is revised, paragraph (g) is redesignated as paragraph (h), and a new paragraph (g) is added to read as follows:

§ 50.34 Contents of applications; technical information.

(a) * * *

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

* * * * *

(g) *Combustible gas control.* All applicants for a reactor construction permit or operating license under this part, and all applicants for a reactor design approval, design certification, or license under part 52 of this chapter, whose application was submitted after October 16, 2003, shall include the analyses, and the descriptions of the equipment and systems required by § 50.44 as a part of their application.

* * * * *

■ 3. Section 50.44 is revised to read as follows:

§ 50.44 Combustible gas control for nuclear power reactors.

(a) *Definitions.*

(1) *Inerted atmosphere* means a containment atmosphere with less than 4 percent oxygen by volume.

(2) *Mixed atmosphere* means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

(b) *Requirements for currently-licensed reactors.* Each boiling or pressurized water nuclear power reactor with an operating license on October 16, 2003, except for those facilities for which the certifications required under § 50.82(a)(1) have been submitted, must comply with the following requirements, as applicable:

(1) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere.

(2) *Combustible gas control.* (i) All boiling water reactors with Mark I or Mark II type containments must have an inerted atmosphere.

(ii) All boiling water reactors with Mark III type containments and all pressurized water reactors with ice condenser containments must have the capability for controlling combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume) so that there is no loss of containment structural integrity.

(3) *Equipment Survivability.* All boiling water reactors with Mark III containments and all pressurized water reactors with ice condenser containments that do not rely upon an inerted atmosphere inside containment to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume).

(4) *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Analyses.* Each holder of an operating license for a boiling water reactor with a Mark III type of containment or for a pressurized water reactor with an ice condenser type of containment, shall perform an analysis that:

(i) Provides an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75 percent of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include consideration of hydrogen control measures as appropriate;

(ii) Includes the period of recovery from the degraded condition;

(iii) Uses accident scenarios that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core.

(iv) Supports the design of the hydrogen control system selected to meet the requirements of this section; and,

(v) Demonstrates, for those reactors that do not rely upon an inerted atmosphere to comply with paragraph (b)(2)(ii) of this section, that:

(A) Containment structural integrity is maintained. Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff in accordance with § 50.90. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on; and

(B) Systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including local detonations, unless such detonations can be shown unlikely to occur.

(c) *Requirements for future water-cooled reactor applicants and licensees.*[2] The requirements in this paragraph apply to all water-cooled reactor construction permits or operating licenses under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after October 16, 2003.

(1) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents.

(2) *Combustible gas control.* All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

(3) *Equipment Survivability.* Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

(4) *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

(ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning.

(5) *Structural analysis.* An applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

(d) *Requirements for future non water-cooled reactor applicants and licensees and certain water-cooled reactor applicants and licensees.* The requirements in this paragraph apply to all construction permits and operating licenses under this part, and to all design approvals, design certifications, combined licenses, or manufacturing licenses under part 52 of this chapter, for non water-cooled reactors and water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this section, any of which are issued after October 16, 2003. Applications subject to this paragraph must include:

(1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and

(2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common defense and security.

■ 4. Section 50.46a is added to read as follows:

§ 50.46a Acceptance criteria for reactor coolant system venting systems.

Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. High point vents are not required for the tubes in U-tube steam generators. Acceptable venting systems must meet the following criteria:

(a) The high point vents must be remotely operated from the control room.

(b) The design of the vents and associated controls, instruments and power sources must conform to appendix A and appendix B of this part.

(c) The vent system must be designed to ensure that:

(1) The vents will perform their safety functions; and

(2) There would not be inadvertent or irreversible actuation of a vent.

PART 52—EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER PLANTS

■ 5. The authority citation for Part 52 continues to read as follows:

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846).

■ 6. In § 52.47, paragraph (a)(1)(ii) is revised to read as follows:

§ 52.47 Contents of applications.

(a) * * *

(1) * * *

(ii) Demonstration of compliance with any technically relevant portions of the

[2] The requirements of this paragraph apply only to water-cooled reactor designs with characteristics (*e.g.*, type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of October 16, 2003.

Three Mile Island requirements set forth in 10 CFR 50.34(f) except paragraphs (f)(1)(xii), (f)(2)(ix) and (f)(3)(v);

* * * * *

Dated at Rockville, Maryland, this 10th day of September 2003.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 03–23554 Filed 9–15–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 72

RIN 3150–AG93

Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations and Monitored Retrievable Storage Installations

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its licensing requirements for dry cask modes of storage of spent nuclear fuel, high-level radioactive waste, and power reactor-related Greater than Class C (GTCC) waste in an independent spent fuel storage installation (ISFSI) or in a U.S. Department of Energy (DOE) monitored retrievable storage installation (MRS). These amendments update the seismic siting and design criteria, including geologic, seismic, and earthquake engineering considerations. The final rule allows the NRC and its licensees to benefit from experience gained in the licensing of existing facilities and to incorporate rapid advancements in the earth sciences and earthquake engineering. The amendments make the NRC regulations that govern certain ISFSIs and MRSs more compatible with the 1996 amendments that addressed uncertainties in seismic hazard analysis for nuclear power plants. The amendments allow certain ISFSI or MRS applicants to use a design earthquake level commensurate with the risk associated with an ISFSI or MRS.

EFFECTIVE DATE: This final rule is effective on October 16, 2003.

FOR FURTHER INFORMATION CONTACT: Keith K. McDaniel, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: (301) 415–5252, e-mail: *kkm@nrc.gov*.

SUPPLEMENTARY INFORMATION:

I. Background

In 1980, the NRC added 10 CFR part 72 to its regulations to establish licensing requirements for the independent storage of spent nuclear fuel and high-level radioactive waste (HLW) (45 FR 74693; November 12, 1980). In 1988, the NRC amended part 72 to provide for licensing the storage of spent nuclear fuel and HLW in an MRS (53 FR 31651; August 19, 1988). Subpart E of Part 72 contains siting evaluation factors that must be investigated and assessed with respect to the siting of an ISFSI or MRS, including a requirement for evaluation of geological and seismological characteristics. ISFSI and MRS facilities are designed and constructed for the interim storage of spent nuclear fuel that has aged for at least one year, other solidified radioactive materials associated with spent fuel storage, and power reactor-related GTCC waste, that are pending shipment to a high-level radioactive waste repository or other disposal site.

The original regulations envisioned ISFSI and MRS facilities as spent fuel pools or single, massive dry storage structures. The regulations required seismic evaluations equivalent to those for a nuclear power plant (NPP) when the ISFSI or MRS is located west of the Rocky Mountain Front (west of approximately 104° west longitude), referred to hereafter as the western U.S., or in areas of known seismic activity east of the Rocky Mountain Front (east of approximately 104° west longitude), referred to hereafter as the eastern U.S. A seismic design requirement, equivalent to the requirements for an NPP (appendix A to 10 CFR part 100) seemed appropriate for these types of facilities, given the potential accident scenarios. For those sites located in the eastern U.S., and not in areas of known seismic activity, the regulations allowed for less stringent alternatives.

For other types of ISFSI or MRS designs, the regulation required a site-specific investigation to establish site suitability commensurate with the specific requirements of the proposed ISFSI or MRS. The NRC explained that for ISFSIs which do not involve massive structures, such as dry storage casks and canisters, the required design earthquake will be determined on a case-by-case basis until more experience is gained with the licensing of these types of units (45 FR 74697).

For sites located in either the western U.S. or in areas of known seismic activity in the eastern U.S., the regulations in 10 CFR part 72 currently require the use of the procedures in appendix A to part 100 for determining the design basis vibratory ground motion at a site. appendix A requires the use of ''deterministic'' approaches in the development of a single set of earthquake sources. The applicant develops for each source a postulated earthquake to be used to determine the ground motion that can affect the site, locates the postulated earthquake according to prescribed rules, and then calculates ground motions at the site.

Advances in the sciences of seismology and geology, along with the occurrence of some licensing issues not foreseen in the development of appendix A to part 100, have caused a number of difficulties in the application of this regulation. Specific problematic areas include the following:

1. Because the deterministic approach does not explicitly recognize uncertainties in geoscience parameters, probabilistic seismic hazard analysis (PSHA) methods were developed that allow explicit expressions for the uncertainty in ground motion estimates and provide a means for assessing sensitivity to various parameters. Appendix A to part 100 does not allow this application.

2. The limitations in data and geologic/seismic analyses, and the rapid evolution in geosciences have required considerable latitude in technical judgment. The inclusion of detailed geoscience assessments in Appendix A has inhibited the use of needed judgment and flexibility in applying basic principles to new situations; and

3. Various sections of Appendix A are subject to different interpretations. For example, there have been differences of opinion and differing interpretations among experts as to the largest earthquakes to be considered and ground motion models to be used, thus often making the licensing process less predictable.

In 1996, the NRC amended 10 CFR parts 50 and 100 to update the criteria used in decisions regarding NPP siting, including geologic and seismic engineering considerations for future NPPs (61 FR 65157; December 11, 1996). The amendments added a new § 100.23 requiring that the uncertainties associated with the determination of the Safe Shutdown Earthquake Ground Motion (SSE) be addressed through an appropriate analysis, such as a PSHA or suitable sensitivity analyses in lieu of appendix A to part 100. This approach takes into account the problematic areas identified above in the earlier siting requirements and is based on developments in the technical field over the past two decades. Further, regulatory guides have been used to address implementation issues. For example, the NRC provided guidance for NPP license applicants in Regulatory Guide 1.165, ''Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion,'' and Standard Review Plan NUREG–0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors,'' Section 2.5.2, ''Vibratory Ground Motion,'' Revision 3. However, the NRC left appendix A to part 100 in place to preserve the licensing basis for existing plants and confined the applicability of § 100.23 to new NPPs.

The NRC is now amending 10 CFR part 72 to require applicants at some locations to address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, for determining the design earthquake ground motion (DE). The use of a probabilistic approach or suitable sensitivity analyses to siting parallels the change made to 10 CFR part 100.

In comparison with an NPP, an operating dry cask ISFSI or MRS facility storing spent nuclear fuel is a passive facility in which the primary activities are waste receipt, handling, and storage. An ISFSI or MRS facility does not have the variety and complexity of active systems necessary to support safe operations at an NPP. Further, the robust cask design required for non-seismic considerations (*e.g.*, drop event, shielding), assure low probabilities of failure from seismic events. In the unlikely occurrence of a radiological release as a result of a seismic event, the radiological consequences to workers and the public are significantly lower than those that could arise at an NPP. The conditions required for release and dispersal of significant quantities of radioactive material, such as high temperatures or pressures, are not present in an ISFSI or MRS. This is primarily due to the low heat-generation rate of spent fuel that has undergone more than one year of decay before storage in an ISFSI or MRS, and to the low inventory of volatile radioactive materials readily available for release to the environment. The long-lived nuclides present in spent fuel are tightly bound in the fuel materials and are not readily dispersible. Short-lived volatile nuclides, such as I–131, are no longer present in aged spent fuel. Furthermore, even if the short-lived nuclides were present during a fuel assembly rupture, the canister surrounding the fuel assemblies is designed to confine these nuclides.

The standards in part 72 Subparts E, ''Siting Evaluation Factors,'' and F, ''General Design Criteria,'' ensure that the dry cask storage designs are very rugged and robust. The casks must maintain structural integrity during a variety of postulated non-seismic events, including cask drops, tip-over, and wind driven missile impacts. These non-seismic events challenge cask integrity significantly more than seismic events. Therefore, the casks have substantial design margins to withstand forces from a seismic event greater than the design earthquake.

Hence, the seismically induced risk from the operation of an ISFSI or MRS is less than at an operating NPP. As a result, the NRC is revising the DE requirements for ISFSI and MRS facilities from the current part 72 requirements, which are equivalent to the SSE for an NPP.

As an additional minor change, the NRC is modifying § 72.212(b)(2)(i)(B) to require general licensees to evaluate dynamic loads, in addition to static loads, in the design of cask storage pads and areas for ISFSIs, to ensure that casks are not placed in unanalyzed conditions. Accounting for dynamic loads in the analysis of ISFSI pads and areas will ensure that pads continue to support the casks during seismic events. General licensees currently evaluate dynamic loads for evaluating the casks, pads and areas, to meet the cask design bases in the Certificate of Compliance, as required by § 72.212(b)(2)(i)(A). Therefore, the rule will not actually require any general licensees operating an ISFSI to re-perform any written evaluations previously undertaken. Specific licensees are currently required, under § 72.122(b)(2), to design ISFSIs to withstand the effects of dynamic loads, such as earthquakes and tornados.

The NRC published the proposed rule, ''Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations and Monitored Retrievable Storage Installations'' in the **Federal Register** on July 22, 2002 (67 FR 47745) for public comment. The NRC stated on September 5, 2002 (67 FR 56876) that it intended to extend the comment period for an additional 15 days to allow interested persons additional time to provide meaningful comments. The public comment period expired on October 22, 2002.

The NRC received nine comment letters on the proposed rule. These comments and the NRC responses are discussed in Section VI of this document, ''Summary of Public Comments on the Proposed Rule.''

II. Objectives

An ISFSI is designed, constructed, and operated under a part 72 specific or general license. A part 72 specific license for an ISFSI is issued to a named person upon application filed under part 72 regulations. A part 72 general license for an ISFSI is issued under 10 CFR 72.210 to persons authorized to possess an NPP license under part 50, without filing a part 72 license application. A general licensee is required to meet the conditions specified in subpart K of part 72. An MRS may be designed, constructed, and operated by DOE under a part 72 specific license.

The final rule reflects changes that are intended to (1) provide benefit from the experience gained in applying the existing regulation and from research; (2) provide needed regulatory flexibility to incorporate into licensing state-of-the-art improvements in the geosciences and earthquake engineering; and (3) make the regulations more risk informed, consistent with the Commission's recent policy.

The objectives of this final rule are to:

1. Require a new specific-license applicant for a dry cask storage facility located in either the western U.S. or in areas of known seismic activity in the eastern U.S., and not co-located with an NPP, to address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, for determining the DE. All other new specific-license applicants for dry cask storage facilities will have the option of complying with the requirement to use a PSHA or suitable sensitivity analyses to address uncertainties in seismic hazard analysis, or other options compatible with the existing regulation. (§ 72.103)

2. Allow new ISFSI or MRS specific-license applicants using a PSHA to select a DE appropriate for and commensurate with the risk associated with an ISFSI or MRS; and

3. Require general licensees to design cask storage pads and areas to adequately account for dynamic loads, in addition to static loads. (§ 72.212)

III. Applicability

This section clarifies the applicability of the new § 72.103 for Part 72 specific licensees, and modified § 72.212(b)(2)(i)(B) for Part 72 general licensees.

Applicability of New § 72.103

(1) Applicants who apply on or after the effective date of the final rule, for a part 72 specific license for a dry cask storage ISFSI or MRS, located in either the western U.S. or in areas of known seismic activity in the eastern U.S., and not co-located with an NPP, will be required to address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, for determining the DE.

(2) Applicants who apply on or after the effective date of the final rule, for a part 72 specific license for a dry cask storage ISFSI or MRS, located in either the western U.S. or in areas of known seismic activity in the eastern U.S., and co-located with an NPP, will have the option of addressing uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, or using the existing design criteria for the NPP, for determining the DE. When the existing design criteria for the NPP are used for an ISFSI at a site with multiple NPPs, the criteria for the most recent NPP must be used.

(3) Applicants who apply on or after the effective date of the final rule, for a part 72 specific license for a dry cask storage ISFSI or MRS, located in the eastern U.S., except in areas of known seismic activity, will have the option of addressing uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, or using a standardized DE described by an appropriate response spectrum anchored at 0.25 g (subject to the conditions in new § 72.103(a)(1)), or using the existing design criteria for the most recent NPP (if applicable), for determining the DE.

(4) The new § 72.103 is not applicable to a general licensee at an existing NPP operating an ISFSI under a part 72 general license anywhere in the U.S.

The changes apply to the design basis of both a dry cask storage type ISFSI and MRS, because these facilities are similar in design. The NRC does not intend to revise the 10 CFR part 72 geological and seismological criteria as they apply to wet modes of storage because applications for this means of storage are not expected and it is not cost-effective to allocate resources to develop the technical bases for such an expansion of the rulemaking. The NRC also does not intend to revise the 10 CFR part 72 geological and seismological criteria as they apply to dry modes of storage that do not use casks because of the lack of experience in licensing these types of facilities.

The applicability of § 72.103 is summarized in the table below.

Applicability of Amended § 72.212(b)(2)(i)(B)

The changes in § 72.212(b)(2)(i)(B), regarding the evaluation of dynamic loads for the design of cask storage pads and areas, will apply to all general licensees for an ISFSI.

The applicability of the modified § 72.212(b)(2)(i)(B) is summarized in the table below.

SUMMARY OF APPLICABILITY

[Design Earthquake Ground Motion for ISFSI or MRS Specific-License Applicants for Dry Cask Modes of Storage on or after the Effective Date of the Final Rule]

Site condition	Specific-license applicant [1]
Western U.S., or areas of known seismic activity in the eastern U.S., not co-located with NPP.	Must use PSHA or suitable sensitivity analyses to account for uncertainties in seismic hazards inevaluations [2].
Western U.S., or areas of known seismic activity in the eastern U.S., and co-located with NPP.	PSHA or suitable sensitivity analyses to account for uncertainties in seismic hazards evaluations [2], or existing NPP design criteria (multi-unit sites—use and co-located withthe most recent criteria). NPP
Eastern U.S., and not in areas of known seismic activity	PSHA or suitable sensitivity analyses to account for uncertainties in seismic hazards evaluations,[2] or existing NPP design criteria, if applicable (multi-unit sites—use the most recent criteria), or an appropriate response spectrum anchored at 0.25g (subject to the conditions in new § 72.103(a)(1)).

[1] New § 72.103 does not apply to general licensees. General licensees must satisfy the conditions specified in 10 CFR 72.212.

[2] Regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

IV. Discussion

The NRC is amending certain sections of part 72 dealing with seismic siting and design criteria for a dry cask ISFSI or MRS. The NRC intends to leave the present § 72.102 in place to preserve the ISFSI licensing bases for applications before the effective date of the rule, and continue the present ISFSI or MRS licensing bases for applications for other than dry cask modes of storage. The NRC is changing the heading of § 72.102, adding a new § 72.103, and modifying § 72.212(b)(2)(i)(B).

A. Change to 10 CFR 72.102

The heading of § 72.102 will be changed to clarify that the present requirements are applicable to ISFSI or MRS specific licensees or specific-license applicants before the effective date of the rule. The requirements of § 72.102 that applied to ISFSI or MRS licensees, or license applicants for other than dry cask modes of storage will continue to apply.

B. New 10 CFR 72.103

New § 72.103 describes the seismic requirements for new specific-license applicants for dry cask storage at an ISFSI or MRS.

1. Remove Detailed Guidance From the Regulation

Part 72 currently requires license applicants for an ISFSI or MRS, in the western U.S. or in other areas of known seismicity, to comply with appendix A to part 100. Appendix A contains both requirements and guidance on how to satisfy those requirements. For example, Section IV, ''Required Investigations,'' of Appendix A states that investigations are required for vibratory ground motion, surface faulting, and seismically induced floods and water waves. Appendix A then provides detailed guidance on what constitutes an acceptable investigation. A similar situation exists in Section V, ''Seismic and Geologic Design Bases,'' of appendix A to part 100.

Geoscience assessments require considerable latitude in judgment because of (a) limitations in data; (b) changing state-of-the-art of geologic and seismic analyses; (c) rapid accumulation of knowledge; and (d) evolution in geoscience concepts. The NRC recognized the need for latitude in judgment when it amended part 100 in 1996.

However, specifying geoscience assessments in detail in a regulation has created difficulty for applicants and the NRC by inhibiting needed latitude in judgment. It has inhibited the flexibility needed in applying basic principles to new situations and the use of evolving methods of analyses (for instance, probabilistic) in the licensing process.

The NRC is adding a new section in part 72 that will provide specific siting requirements for an ISFSI or MRS instead of referencing another part of the regulations. The amended regulation will also reduce the level of detail by placing only basic requirements in the rule and providing the details on methods acceptable for meeting the requirements in an accompanying guidance document. Thus, the revised regulation contains requirements to:

(i) Evaluate the geological, seismological, and engineering characteristics of the proposed site;

(ii) Establish a DE; and

(iii) Identify the uncertainties associated with these requirements.

Detailed guidance on the procedures acceptable to the NRC for meeting the requirements are provided in Regulatory Guide 3.73, ''Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations.''

2. Address Uncertainties and Use Probabilistic Methods

The existing approach for determining a DE for an ISFSI or MRS, embodied in Appendix A to Part 100, relies on a ''deterministic'' approach. Using this deterministic approach, an applicant develops a single set of earthquake sources, develops for each source a postulated earthquake to be used as the source of ground motion that can affect the site, locates the postulated earthquake according to prescribed rules, and then calculates ground motions at the site.

Although this approach has worked reasonably well for the past several decades in the sense that the SSE for NPPs sited with this approach are judged to be suitably conservative, the approach has not explicitly recognized uncertainties in geosciences parameters. Because so little is known about earthquake phenomena (especially in the eastern U.S.), there have been differences of opinion and differing interpretations among experts as to the largest earthquakes to be considered and ground-motion models to be used, often making the licensing process less predictable.

Probabilistic methods that have been developed in the past 15 to 20 years for evaluation of seismic safety of nuclear facilities allow explicit incorporation of different models for zonation, earthquake size, ground motion, and other parameters. The advantage of using these probabilistic methods is their ability to incorporate different models and data sets, thereby providing an explicit expression for the uncertainty in the ground motion estimates and a means of assessing sensitivity to various input parameters. The western and eastern U.S. have fundamentally different tectonic environments and histories of tectonic deformation. Consequently, application of these probabilistic methodologies has revealed the need to vary the fundamental PSHA methodology depending on the tectonic environment of the site.

In 1996, when the NRC accepted the use of a PSHA methodology or suitable sensitivity analyses in § 100.23, it recognized that the uncertainties in seismological and geological information must be formally evaluated and appropriately accommodated in the determination of the SSE for seismic design of NPPs. The NRC further recognized that the nature of uncertainty and the appropriate approach to account for it depends on the tectonic environment of the site and on properly characterizing parameters input to the PSHA. Methods other than probabilistic methods (PSHA), such as sensitivity analyses, may be adequate for some sites to account for uncertainties. The NRC believes that certain new applicants for ISFSI or MRS specific licenses, as described in Section III, ''Applicability,'' of this document, must use probabilistic methods or other sensitivity analyses to account for uncertainties instead of using Appendix A to Part 100. The NRC does not intend to require new ISFSI or MRS specific-license applicants that are co-located with an NPP to address uncertainties because the criteria used to evaluate existing NPPs are considered to be adequate for ISFSIs, in that the criteria have been determined to be safe for NPP licensing, and the seismically induced risk of an ISFSI or MRS is considerably lower than that of an NPP, as described in Section IV of this document.

The key elements of the NRC's approach for seismic and geologic siting for ISFSI or MRS license review and approval consists of:

a. Conducting site-specific and regional geoscience investigations;

b. Setting the target exceedance probability commensurate with the level of risk associated with an ISFSI or MRS;

c. Conducting PSHA and determining ground motion level corresponding to the target exceedance probability;

d. Determining if other sources of information change the available probabilistic results or data for the site; and

e. Determining site-specific spectral shape, and scaling this shape to the ground motion level determined above.

In addition, the NRC will review the application using all available data including insights and information from previous licensing experience. Thus, the revised approach requires thorough regional and site-specific geoscience investigations. Results of the regional and site-specific investigations must be considered in applying the probabilistic method. Two current probabilistic methods are the NRC-sponsored study conducted by Lawrence Livermore National Laboratory and the Electric Power Research Institute's seismic hazard study. These are essentially regional studies. The regional and site-specific investigations provide detailed information to update the database of the hazard methodology to make the probabilistic analysis site-specific.

Applicants must also incorporate local site geological factors, such as stratigraphy and topography, and account for site-specific geotechnical properties in establishing the DE. Guidelines to incorporate local site factors and advances in ground motion attenuation models, and to determine ground motion estimates, are outlined in NUREG–0800, Section 2.5.2.

Methods acceptable to the NRC for implementing the revised regulation related to the PSHA or suitable sensitivity analyses are described in RG 3.73.

3. Revise the Design Earthquake Ground Motion

The present DE in part 72 is based on the deterministic requirements contained in Appendix A to 10 CFR Part 100 for NPPs. In the Statement of Considerations accompanying the initial part 72 rulemaking, the NRC recognized that the required design earthquake need not be as high as for an NPP and should be determined on a ''case-by-case'' basis until ''more experience is gained with licensing of these types of units'' (45 FR 74697; November 12, 1980). With the advances in probabilistic seismic hazard evaluation techniques, over 10 years of experience in licensing dry cask storage (10 specific licenses have been issued and 9 locations use the general license provisions), and analyses demonstrating robust behavior of dry cask storage systems (DCSSs) in accident scenarios, the NRC now has a reasonable basis to consider more appropriate DE parameters for a dry cask ISFSI or MRS. Therefore, in those instances when an ISFSI or MRS specific-license applicant uses PSHA methods, the NRC will allow a DE commensurate with the lower risk associated with these facilities.

I. Factors that result in the lower radiological risk at an ISFSI or MRS compared to an NPP include the following:

a. In comparison with an NPP, an operating ISFSI or MRS is a passive facility in which the primary activities are waste receipt, handling, and storage. An ISFSI or MRS does not have the variety and complexity of active systems necessary to support an operating NPP. After the spent fuel is in place, an ISFSI or MRS is essentially a static operation.

b. During normal operations, the conditions required for the release and dispersal of significant quantities of radioactive materials are not present. There are no components carrying fluids at high temperatures or pressures during normal operations or under design basis accident conditions to cause the release and dispersal of radioactive materials. This is primarily due to the low heat-generation rate of spent fuel that has undergone more than one year of decay before storage in an ISFSI or MRS, and to the low inventory of volatile radioactive materials readily available for release to the environment.

c. The long-lived nuclides present in spent fuel are tightly bound in the fuel materials and are not readily dispersible. Short-lived volatile nuclides, such as I–131, are no longer present in aged spent fuel. Furthermore, even if the short-lived nuclides were present during a fuel assembly rupture, the canister surrounding the fuel assemblies would confine these nuclides. Therefore, the NRC believes that the seismically induced radiological risk associated with an ISFSI or MRS is significantly less than the risk associated with an NPP.

II. Additional rationale for allowing the use of a DE level commensurate with the risk associated with an ISFSI or MRS includes the following:

a. Because the DE is defined as a smooth broad-band spectrum, which envelops the controlling earthquake responses, the vibratory ground motion specified is conservative.

b. To evaluate dry cask storage systems' behavior during an earthquake, typical storage systems (one a cylindrical cask, HI-STORM 100, the other a concrete module type, NUHOMS) were analyzed for a range of earthquakes. Based on the results of the analyses, the NRC has concluded that a free-standing dry storage cask remains stable and will not tip-over, or would not slide and impact the adjacent casks during an earthquake approximately equal to the magnitude of a SSE for an NPP. Additionally, parametric studies indicated that dry cask storage systems have significant margins against tip-over and sliding, to withstand an earthquake significantly higher in magnitude than the SSE for an NPP, without releasing radioactivity. Further, a cask is analyzed for a non-mechanistic tip-over event during an earthquake, to verify that it would maintain its structural integrity, and radioactivity from spent fuel would not be released to the environment. Therefore, based on drop accident analyses and non-mechanistic tip-over event evaluations, and on the results of the generic studies for the cask behavior during an earthquake, it can be concluded that there would be no radiological consequences at a dry cask ISFSI or MRS facility due to an earthquake.

c. The rational for allowing a DE for an ISFSI or MRS to be lower than a DE for an NPP is consistent with the approach used in DOE Standard DOE–STD–1020, ''Natural Phenomena Hazards Design Evaluation Criteria for Department of Energy Facilities.''

Regulatory Guide 3.73 (formerly DG–3021) recommends an acceptable mean annual probability of exceedance (MAPE) for the DE that is commensurate with the lower risk associated with an ISFSI or MRS as compared to an NPP. The basis for the recommendation is provided in a report entitled, ''Selection of the Design Earthquake Ground Motion Reference Probability''. This report may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr@nrc.gov.* Discussion on the recommended mean annual probability of exceedance is also in Section VI of this FRN, ''Summary of Public Comments on the Proposed Rule.''

C. Change to 10 CFR 72.212(b)(2)(i)(B)

The NRC is modifying § 72.212(b)(2)(i)(B) to require that general licensees evaluate dynamic loads, in addition to static loads, in the design of cask storage pads and areas for ISFSIs to ensure that casks are not placed in unanalyzed conditions. During a seismic event, the cask storage pads and areas experience dynamic loads in addition to static loads. The dynamic loads depend on the interaction of the casks, cask storage pads, and areas. Consideration of the dynamic loads of the stored casks, in addition to the static loads, for the design of the cask storage pads and areas, will ensure that the cask storage pads and areas will perform satisfactorily during a seismic event.

The revision will also require consideration of potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motions. Depending on the properties of soil and structures, the free-field earthquake acceleration input loads may be amplified at the top of the storage pad. These amplified acceleration input values must be bound by the design bases seismic acceleration values for the cask, specified in the Certificate of Compliance. Liquefaction of the soil and instability during vibratory motion due to an earthquake may affect the cask stability.

The changes to § 72.212 will not actually impose a new burden on the general licensees because they currently need to consider dynamic loads to meet the requirements in § 72.212(b)(2)(i)(A). Section 72.212(b)(2)(i)(A) requires that general licensees perform written evaluations to meet conditions set forth in the cask Certificate of Compliance. These Certificates of Compliance require that dynamic loads, such as seismic and tornado loads, be evaluated to meet the cask design bases. Specific licensees are currently required, under § 72.122(b)(2), to design ISFSIs to withstand the effects of dynamic loads, such as earthquakes and tornados.

V. Related Regulatory Guide and Standard Review Plans

On July 22, 2002, the NRC published DG–3021, ''Site Evaluations and Determination of Design Earthquake Ground Motion for Seismic Design of Independent Spent Fuel Storage Installations and Monitored Retrievable Storage Installations'' for public comment (67 FR 48956; July 26, 2002). Regulatory Guide 3.73, *Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations* (formerly DG–3021), provides guidance to licensees for procedures acceptable to the NRC staff for:

(1) Conducting a detailed evaluation of site area geology and foundation stability;

(2) Conducting investigations to identify and characterize uncertainty in seismic sources in the site region important for the probabilistic seismic hazard analysis (PSHA);

(3) Evaluating and characterizing uncertainty in the parameters of seismic sources;

(4) Conducting PSHA for the site; and

(5) Determining the DE to satisfy the requirements of 10 CFR Part 72.

This guide describes acceptable procedures and provides a list of references that present acceptable methodologies to identify and characterize capable tectonic sources and seismogenic sources. Section IV.B of this SUPPLEMENTARY INFORMATION describes the key elements of the regulatory guide. A document announcing the availability of Regulatory Guide 3.73 will be published in the **Federal Register** in the near future.

Requests for single copies of active regulatory guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future guides should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section, or by fax to (301) 415–2289; email *distribution@nrc.gov.* Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike (first floor), Rockville, MD; the PDR's mailing address is U.S. NRC PDR, Washington, DC 20555; telephone (301) 415–4737 or 1–(800) 397–4209; fax (301) 415–3548; e-mail *pdr@nrc.gov.*

In the future, editorial changes to NUREG–1536, ''Standard Review Plan for Dry Cask Storage Systems,'' and NUREG–1567, ''Standard Review Plan for Spent Fuel Dry Storage Facilities,'' will be made. For example, the standard review plans will be updated to reference the new § 72.103 and Regulatory Guide 3.73.

VI. Summary of Public Comments on the Proposed Rule

This section presents a summary of the public comments received on the proposed rule and supporting documents, the NRC's response to the comments, and changes made in the final rule and supporting documents as a result of these comments.

The NRC received nine comment letters on the proposed rule from eight commenters. The commenters were the Nuclear Energy Institute (NEI), the U.S. Department of Energy (DOE), two nuclear power utilities, three State agencies, and one license applicant for an independent spent fuel storage installation. All the commenters agreed with the proposal to address uncertainty by requiring the use of a PSHA or suitable sensitivity analyses for an ISFSI or MRS in the western U.S., not co-located with an NPP, and in areas of known seismic activity in the eastern U.S. However, commenters were divided on the specific question for public comment related to the appropriate value for the MAPE posed by the Commission in the proposed rule. These comments are summarized in this section under the heading ''Related Regulatory Guide.'' All commenters supported the concept of requiring general licensees to evaluate both dynamic loads and static loads for ISFSI and MRS cask storage pads and areas.

Copies of the public comments are available for review in the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD. A review of the comments and the NRC responses follow:

General Comments

Comment 1: A commenter stated that proposed 10 CFR 72.103(f)(1) does not comply with the notice and comment requirements of Section 553 of the Administrative Procedure Act (APA) because of the way the rule is structured. The commenter believes that the proposed rule ''is in the guise of a substantive rule,'' but that the substantive requirements are found in the draft guidance, a document which is not a rule. In the commenter's view, ''the Commission attempts to give concrete form to its proposed rule through an interpretative document, DG–3021, and the Commission thereby circumvents [APA] § 553 notice and comment rulemaking procedures,'' citing *Paralyzed Veterans of America* v. *D.C. Arena L.P.,* 117 F.3d 579 (D.C. Cir. 1997). According to the commenter, a significant defect of this structure is that the rule gives no standards against which a licensing board or intervenors may evaluate whether an applicant has complied with the rule and, instead, gives ''unbridled and unchecked discretion to the staff in determining the seismic design standard for ISFSIs sited in seismic areas.'' The proposed rule, in the commenter's view, has no force of law because it has no binding standards and thus is unenforceable. Another commenter disagreed and supported the NRC's view that the rule is substantive and in compliance with the APA.

Response: First, the NRC rejects the claim that the rule is not being promulgated in compliance with § 553 of the APA. Section 553 requires that notice of a proposed rulemaking be published in the **Federal Register**, including the terms or substance of the proposed rule, and that interested persons be given an opportunity to comment. The APA also provides an exception for interpretative rules and general statements of policy enabling those documents to be issued as final rules without prior notice and comment. In this case, the NRC has not availed itself of the exception but rather has issued both the draft guidance and the proposed rule for public comment. Thus, there has been no violation of the notice and comment requirements of Section 553 of the APA even if the guidance were to be considered part of the rule. The *Paralyzed Veterans* case, cited by the petitioner, concerned a guidance document issued by the Department of Justice which had been issued *without* prior notice and comment and raised the issue whether the Government could rely upon the guidance in an enforcement action. The court ultimately found that there was no need for the Government to rely on the guidance to enforce the regulation. Here, the guidance has been issued for comment and the NRC does not contend, as explained below, that the guidance is legally enforceable.

Second, the NRC does not agree that ''substantive requirements'' have been placed in the guidance document. Regulatory Guide 3.73 (formerly DG–3021) provides information on methods acceptable to the NRC for implementing specific parts of the rule, but it does not place any particular requirements on applicants. As the commenter points out, ''staff regulatory guides are not regulations, do not have the force of regulations, and when challenged, are considered only one way in which an applicant may meet the regulations.''

Finally, the commenter really appears to be objecting to the NRC's risk-informed, performance-based approach in this rulemaking in lieu of the deterministic approach for determining a design earthquake embodied in Appendix A to 10 CFR Part 100. The overall performance criteria for protection against environmental conditions and natural phenomena in the design of Part 72 facilities are contained in 10 CFR 72.122(b) of the NRC's regulations. In particular, § 72.122(b)(2)(i) provides:

Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes * * * without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect:

(A) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated; and

(B) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena.

These performance criteria are supplemented by the requirements of 10 CFR 72.103 governing selection of a site and determination of a DE. This new regulation provides specific siting requirements for an ISFSI or MRS instead of referencing another part of the regulations (Appendix A to Part 100). This new regulation also reduces the level of detail by placing only basic requirements in the rule and providing the details on methods acceptable for meeting the requirements in an accompanying guidance document. Thus, the new 10 CFR 72.103(f) establishes basic requirements for determining a DE for use in the design of structures, systems, and components of the ISFSI or MRS. These regulations include a requirement that the geological, seismological, and engineering characteristics of a proposed site and its environs be investigated in sufficient scope and detail to provide sufficient information to support evaluations performed to arrive at estimates of the DE (§ 72.103(f)(1)); a requirement that a DE be determined for the site (§ 72.103(f)(2)); and a requirement that uncertainties be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis or suitable sensitivity analyses (§ 72.103(f)(2)(i)). The regulation further requires determinations of the potential for surface tectonic and nontectonic deformations (§ 72.103(f)(2)(ii)); the design bases for seismically induced floods and water waves (§ 72.103(f)(2)(iii)); and the siting factors for other design conditions, such as liquefaction potential (§ 72.103(f)(2)(iv)), as well as a requirement that the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum (§ 72.103(f)(3)). More specific guidance for meeting these standards, including guidance on an acceptable reference probability, is provided in Regulatory Guide 3.73 (formerly DG–3021).

Determining whether an applicant has complied with these performance standards may be more difficult than would be the case with a prescriptive regulation; however, that does not mean that the NRC has ''unbridled discretion'' in deciding whether the standards are met nor that the standards (as opposed to the guidance) are not binding. The NRC uses informed technical judgment to determine if an application has satisfactorily met the standards. The NRC's rationale and judgment are expressed in a safety evaluation report (SER) subject to evaluation and potential challenge by members of the public. In the event of a hearing, a licensing board would have the technical skills necessary to evaluate any conflicting claims.

Comment 2: A commenter noted that, although the NRC's approach is similar to that used in the amendments issued for seismic evaluation for the siting of NPPs, the NRC has no compelling reason to follow that approach. First, the commenter argued, if the approach violates the APA, it should be rejected. Second, the commenter stated that because no new applications for siting NPPs have been submitted using the new requirements, the rule has not been put to the test. Finally, the commenter indicated that there are no data for ISFSIs that establish design basis ground motions, unlike the SSE for a nuclear power plant, which has at least some data to provide guidance to the NRC and the public.

Response: First, the NRC disagrees that either the amendments issued for the seismic evaluation of siting of NPPs or these Part 72 amendments have been issued in violation of the APA. See comment 1. Second, although no new license applications for siting of NPPs have been received to test the new requirements in 10 CFR § 100.23, the guidance associated with the use of probabilistic methods for siting of NPPs (Regulatory Guide 1.165) has been used in the PSHA prepared for a proposed ISFSI site. It is also being followed by applicants for an early site permit under 10 CFR Part 52. Finally, the NRC agrees that there are limited data for ISFSIs that establish design basis ground motions because the current Part 72 regulations for seismic design of ISFSIs are conservatively based on the nuclear power plant seismic design criteria, and thus, are not risk-informed. However, experience has been gained in the design and construction of numerous facilities using the philosophy of a graded, risk-informed approach described in the standard building codes, similar to the approach proposed in the rule for ISFSIs. The graded risk-informed approach is also used by the Department of Energy in designing its facilities for seismic loads with risks varying from conventional facilities to NPPs.

Comment 3: A commenter noted that if clear seismic standards are not established in the rule, the opportunity for interested persons to participate in a licensing proceeding involving the seismic design of an ISFSI will become essentially prohibited. This is because a panoply of specific expertise is needed to evaluate the seismic design and there is only a small universe of seismic experts. Utilizing these experts is often not feasible because of the financial burden on intervenors in obtaining highly specialized expertise to analyze probabilistic seismic risks and design of nuclear facilities.

Response: The NRC believes the standards for ISFSI or MRS facility earthquake designs are clear. See the response to Comment 1. However, the NRC recognizes that the proposed use of the probabilistic methods in seismic design of ISFSIs is more complex than the current deterministic methods of 10 CFR Part 100 Appendix A, and would require specific expertise to participate in the licensing proceedings. The NRC staff's safety evaluation report (SER) that independently assesses the applicant's method of compliance with regulations is available to assist the public in evaluating the risk of the facility and could help intervenors to focus their resources. The NRC does not intend to limit public participation in the licensing process; however, the Congress has barred the use of appropriated funds to pay the expenses of, or otherwise compensate, parties who intervene in NRC regulatory or adjudicatory proceedings.

Comment 4: A commenter stated that the proposed rule placed too much stock on the integrity of the dry storage cask. The commenter indicated that of the 19 ISFSI licenses issued in the past decade, none were in seismic areas. The NRC has not licensed unanchored cylindrical casks in any seismic areas. The commenter noted that there are no performance data, test data, or earthquake experience data for dry casks or for ISFSIs. The commenter further stated that the rule is based on principles that are antithetical to earthquake engineering principles because, for unanchored casks, the NRC relies solely on the predictions of non-linear computer models. The commenter also stated that, up to this point, the non-linear computer model predictions of the seismic behavior of casks have not been validated with shake table data or actual performance data. The commenter also stated that without adequate and reliable performance and test data, it cannot be determined if the casks will actually provide the critical barrier described and relied upon in the rule. Another commenter stated that non-linear dynamic analyses are inherently reliable. Further, the commenter noted that proper input parameters for cask stability analyses are not elusive unknowns but can be determined from basic physical principles, and that these analyses have been shown not to be highly sensitive to changes in input parameters. Therefore, the commenter argued, shake table testing is unnecessary.

Response: The integrity of the dry storage cask during an earthquake is a key to protecting the health and safety of the public because it confines the radioactivity during a potential accident event, such as an earthquake, and prevents it from being dispersed into the environment. Contrary to traditional building designs, the cask design is not governed by stresses resulting from an earthquake, but is governed by requirements resulting from shielding, thermal, criticality, and postulated handling accidents. Therefore, the critical performance requirement for a cask is that it would remain stable and not displace excessively to impact adjacent casks. The cask stability can be determined by nonlinear dynamic analyses, considering uncertainties in engineering parameters, and using multiple computer codes. The NRC has also performed structural analyses of casks tipping and sliding. In neither case did the canister fail.

It is a common engineering practice to design and build structures, including new design concepts, based on detailed structural analyses using sound engineering principles and laws of physics, without performing confirmatory experiments. For example, new concepts in structural designs and construction of landmark structures, such as the Sears Tower, Hancock Tower, Eiffel Tower, and space vehicles were based solely on analyses.

The advent of computers has helped in the development of analytical tools, including the non-linear dynamic analyses. Results of these analyses are being used to design structures more complex than a dry storage cask. The concept of free-standing casks is not new. The buildings the NRC uses every day are free-standing on a foundation, and thus would move during an earthquake. The analytical tools for non-linear structural analyses are verified and validated using multiple computer codes and available experimental data. Therefore, shake table tests or actual performance data are not necessary.

Comment 5: A commenter requested a rule to establish a definitive design basis earthquake at a return period level [the return period of an earthquake is an inverse of the mean annual probability of exceedance (MAPE) of the earthquake] greater than 2,000 years that is tied to defined risk and performance goals.

Response: The NRC does not agree that we must establish a definitive design basis earthquake by rule. The current regulations in § 72.122(b)(2)(i), require that the structures, systems, and components of an ISFSI or MRS must be designed to withstand the effects of natural phenomena, such as earthquakes, without impairing their capability to perform their intended design functions. For earthquakes, these requirements are then supplemented by the requirements at §§ 72.102, 72.103, and 72.122 for detailed site investigations and appropriate consideration of the most severe of the natural phenomena and associated probability of occurrence, including consideration of uncertainties, in the prediction of earthquakes. This approach is consistent with the NRC's philosophy of using risk-informed, performance-based regulations. In a risk-informed, performance-based approach, the design of the ISFSI or MRS facility is based on an assessment of the radiological risk (potential for adverse consequences) due to an earthquake. Thus, specifying a value for the reference probability in the rule would preclude applicants from considering structures, systems, and components with risks other than the risk associated with the specified reference probability.

Comment 6: A commenter stated that the supplementary information in the final rule should state that the NRC's policy for promulgating risk-informed regulations was a primary motivation for the rule changes.

Response: The NRC agrees that the supplementary information for the final rule should more clearly state that the rule was amended, in part, to conform to the Commission's recent policy to increase the use of risk insights and information in its regulatory applications. An additional statement has been added to Section II, Objectives, of the Supplementary Information portion of this document, that states the intent to revise the regulation in accordance with this policy.

Applicability of Proposed § 72.103

Comment 7: A commenter requested clarification of the proposed rule so that applicants for an ISFSI co-located with an NPP have the option of using the existing DE of the NPP without any further evaluations and that this applies to all sections of the rule. The commenter pointed out that the proposed amendments at §§ 72.103(a)(2) and 72.103(b), as well as explanatory statements made in the proposed rule indicate that applicants for an ISFSI that are co-located with an NPP have the option of using the existing NPP design criteria without additional evaluations, but that this option is not identified in § 72.103(f).

Response: To further clarify the NRC's intent that an applicant for an ISFSI that is co-located with an NPP has the option of using the existing DE of the NPP without the need to undertake any additional evaluations of the sort described in § 72.103(f), the introductory phrase of that section has been modified so that it now reads: ''Except as provided in paragraphs (a)(2) and (b) of this section, the DE for use in the design of structures, systems, and components must be determined as follows.''

Comment 8: Two commenters stated that the criteria presented for establishing the DE for ISFSI and MRS sites at existing NPPs allows for the use of the existing NPP SSE as one alternative. This alternative is key to ensuring that significant new probabilistic ground motion studies are not required at existing NPP sites.

Response: The commenters are correct. The regulatory changes allowing the licensee flexibility to use the existing SSE for an NPP at co-located ISFSIs or MRSs means that new studies are not required at ISFSIs or MRSs co-located with NPPs.

Alternative of Adopting 10 CFR 100.23

Comment 9: One commenter recommended withdrawing the proposed rule and adopting the option of directing new applicants for specific licenses to comply with 10 CFR 100.23 in its entirety, including conforming the DE to the SSE criteria. The commenter noted that by adopting § 100.23 in its entirety, there would be no need to make distinctions among locations of facilities and the rule would incorporate state-of-the-art improvements in the geosciences and earthquake engineering and would allow uncertainty to be addressed. The commenter further noted that NRC had cited its 10 years of experience in reviewing dry cask storage installation applications as a reasonable basis for allowing an exceedance probability greater than that applied to a nuclear power plant, but pointed out that this was 10 years of analytical, not practical experience. In the commenter's view, this lack of practical experience, and the fact that a probabilistic analysis is, by its very nature, risk-informed with respect to uncertainty, means that there does not seem to be a quantifiable safety basis for any exceedance margin other than that now applied to seismic analysis for nuclear power plant proposals. The commenter stated that, absent any definitive experience, the seismic design criteria for an ISFSI should be no less protective than that of a nuclear power plant.

Response: The NRC disagrees that new applicants for specific licenses should comply with § 100.23 in its entirety, including conforming the DE to the SSE criteria. Adopting the recommendation would fail to recognize the differences in risk between an NPP and an ISFSI or MRS facility in seismic design requirements. This is counter to the Commission policy encouraging development of risk-informed, performance-based regulations, and the Commission's Performance Goals.

The NRC acknowledges that actual earthquake performance data for ISFSI facilities are not available and thus that NRC's decision to allow an exceedance probability greater than that applied to a nuclear power plant is not based on practical experience. However, NRC has gained sufficient analytical experience to understand the performance of these facilities, by reviewing the analyses of these facilities performed by the licensees, and by performance of independent analyses. Additionally, experience has been gained in the design and construction of numerous facilities using the philosophy of a risk-informed approach described in the standard building codes, similar to the one proposed in the rule for ISFSIs. The risk-informed approach is also used by the Department of Energy in designing its facilities for seismic loads with risks varying from conventional facilities to NPPs. NRC staff's analyses show that ISFSI storage casks are sufficiently robust, due to design requirements other than for earthquakes, that there is no release of radioactivity at an ISFSI site with a DE at a magnitude equal to the SSE for a NPP. This analytical experience provides a basis for allowing an exceedance probability greater than that applied to a nuclear power plant.

Proposed Change to 10 CFR 72.103

Comment 10: With respect to the provision in § 72.103(b) that sites ''that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided,'' a commenter stated that the definition of ''range of strong near-field ground motion'' is not well defined but is often believed to be about 15 km. The commenter noted that this is a very large set-back from faults. The commenter argued that the key issue is that the design ground motion should represent the conditions at the site. If a site is located close to a large capable fault, then near-fault effects should be incorporated into the design ground motions rather than excluding these site locations.

Response: The NRC agrees with the comment. The sentence: ''Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.'' has been removed from § 72.103(b). Section 72.103(f)(2)(iv) requires an evaluation of the effects of vibratory ground motion that may affect the design and operation of the proposed ISFSI or MRS. Therefore, near-fault effects must be included in the development of the ground motion used in design.

Comment 11: One commenter suggested removing the distinction in § 72.103 between western U.S. and eastern U.S. The commenter stated that the characterization of areas of known seismicity east of the Rocky Mountain Front as including three specific areas is misleading. The commenter argued that the entire region of the U.S. east of the Rocky Mountain Front is subject to earthquake occurrence and that one area should not be treated differently from another for the purpose of assessing seismic sources. Further, the commenter stated that 10 CFR part 100, appendix A, does not allow for less stringent alternatives for any area. Rather, the commenter noted, the fundamental requirements of that regulation apply uniformly to all regions of the U.S., independent of variations in the local rate of seismicity.

Response: In specifying the criteria for determining the DE, the current part 72 regulations distinguish between the western U.S. and the eastern U.S. Although the entire eastern U.S. is subject to earthquake occurrence, the areas east of the Rocky Mountain Front, except in specific areas of known seismic activity, do not experience significant seismic activity. Therefore, the use of an appropriate seismic response anchored at 0.25 g is considered as bounding for the design. However, for the western U.S. there is significant seismic activity varying from region to region. Therefore, it is not practical to use a bounding approach in specifying the DE for those sites.

However, if the applicant chooses the option of performing the PSHA for a site located in the eastern U.S., as allowed in § 72.103(a)(2), the seismic sources are assessed with the same rigor as the seismic sources for the PSHA performed for a site located in the western U.S. (§ 72.103(f)). In this case, the regulatory requirements of assessing the seismic sources for the PSHA method would apply uniformly to all regions of the U.S., independent of variations in the local rate of seismicity.

Comment 12: One commenter suggested inserting the word ''sites'' after ''NY'' in the first sentence of § 72.103(a)(1) to be consistent with language in § 72.102.

Response: The NRC agrees with the commenter's suggestion. The word ''sites'' will be inserted after ''NY'' in the first sentence of § 72.103(a)(1) to be consistent with language in § 72.102. In addition, other minor editorial changes have been made to this sentence.

Remove Detailed Guidance From the Regulation

Comment 13: One commenter stated that removing detailed guidance from the regulation that is related to analyzing non-seismic factors affecting geologic stability of the site would allow excessive discretion for the applicant and would result in too much uncertainty for a safety evaluation. This commenter noted that removing requirements for specific types of evaluation also removes the certainty for both the license applicant and the public as to what is expected during a review. The commenter requested retaining appendix A of part 100 as requirements for licensing.

Response: See the response to Comment 1.

Comment 14: A commenter questioned NRC's statement explaining that NRC proposed to remove detailed guidance from the regulation, in part, because ''specifying geoscience assessments in detail in a regulation has created difficulties for applicants and the NRC by inhibiting needed latitude in judgment [and] [i]t has inhibited the flexibility needed in applying basic principles to new situations.'' This commenter asked for an explanation as to how and when latitude and flexibility in judgment and in applying basic principles to new situations because geoscience assessments were specified in detail in a regulation, were inhibited.

Response: The current regulation (§ 72.102) requires that for areas of known potential seismic activity, seismicity will be evaluated by the techniques of appendix A to part 100. appendix A contains both requirements and guidance on how to satisfy the requirements. For example, Section IV, ''Required Investigations,'' of appendix A, states that investigations are required for vibratory ground motion, surface faulting, and seismically induced floods and water waves. Appendix A then provides detailed guidance on what constitutes an acceptable investigation. Such investigations require considerable latitude in judgment. This latitude in judgment is needed because of limitations in data and rapidly evolving state-of-the-art geologic and seismic analyses.

However, having geoscience assessments detailed and cast in a regulation has created difficulty for applicants and the NRC in terms of inhibiting the use of needed latitude in judgment. Also, it has inhibited flexibility in applying basic principles to new situations and the use of evolving methods of analyses (for instance, probabilistic) in the licensing process.

As an example, a prescriptive requirement of applying the capable fault criteria (see part 100, appendix A, § III(g)) to sites in California meant conducting investigations and analyses for surface rupture potential. If a fault does not cause a surface rupture (blind fault), the fault would not be considered a capable fault under the appendix A criteria, and thus would not be considered in determining the DE. This would lead to seismic hazard at a facility which would be not conservative. This has been demonstrated by the occurrences of the 1989 Loma Prieta, 1992 Petrolia, and 1994 Northridge earthquakes during which the causative faults did not rupture ground surface. On the other hand, the young faults, the last movements of which may satisfy the appendix A criteria for classifying them as capable faults, may not be capable faults in the true meaning of the criteria because the most recent displacements on them may be related to non-tectonic natural phenomena. In this case, use of the appendix A criteria would lead to a finding of seismic hazard at a facility which would be overly conservative. Inclusion of detailed criteria or specific numbers in the regulation prevents a scientific evaluation of methodologies and approaches that advance with the state of the art, and the rule eventually becomes a hindrance to the exercise of rational judgement.

Address Uncertainties and Use Probabilistic Methods

Comment 15: A commenter urged revision of § 72.103 to continue to allow an applicant located in the western U.S. or in areas of known seismic activity in the eastern U.S., and not co-located with an NPP, to use a deterministic analysis similar to the analysis specified in appendix A to 10 CFR part 100, for developing design earthquake ground motions because a utility may decide to perform seismic hazards analysis on deterministic bases that are more conservative than the proposed rule.

Response: In using the deterministic approach for determining a SSE for a nuclear reactor site embodied in appendix A to 10 CFR part 100, there have often been differences of opinion and differing interpretations among experts as to the largest earthquakes to be considered and ground-motion models to be used. This often makes the licensing process relatively unstable. Over the past decade, analysis methods for incorporating these different interpretations have been developed and used. These ''probabilistic'' methods have been designed to allow explicit incorporation of different models for zonation, earthquake size, ground motion, and other parameters. The advantage of using these probabilistic methods is the ability to incorporate different models and different data sets and weight them using judgments as to the validity of the different models and data sets. This process provides an explicit expression for the uncertainty in the ground motion estimates and a means of assessing sensitivity to various input parameters.

Section 72.103 explicitly recognizes that there are inherent uncertainties in establishing the seismic and geologic design parameters and requires the use of a probabilistic seismic hazard methodology capable of propagating uncertainties to address these uncertainties. The rule further recognizes that the nature of uncertainty and the appropriate approach to account for it depend greatly on the tectonic regime and parameters, such as the knowledge of seismic sources, the existence of historical and recorded data, and the understanding of tectonics. Therefore, methods other than the probabilistic methods, such as sensitivity analyses, may be adequate for some sites to account for uncertainties.

Consistent with § 100.23 for an NPP, § 72.103 does not allow the use of the deterministic methods in appendix A to 10 CFR part 100, to determine the DE because the deterministic methods do not account for the uncertainties in the seismic hazard analysis. However, § 72.103 allows the applicant to use methods other than the probabilistic methods, such as sensitivity analyses, to account for uncertainties. Additionally, § 72.103 allows a utility applying for a specific license for an ISFSI co-located at an NPP, the option of using the seismic design criteria of the NPP, which may be based on the deterministic methods of appendix A to 10 CFR part 100.

For these reasons, the NRC declines to amend § 72.103 as suggested by the commenter. However, a utility applying for a specific license for an ISFSI co-located at an NPP has the option of using the seismic design criteria of the NPP.

Comment 16: A commenter stated that the use of the term ''uncertainty'' in the Background section of the proposed rule (67 FR 47746) is ambiguous, and suggested that the term be revised to ''aleatory uncertainty''. The commenter stated that the report ''Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,'' NUREG/CR–6372 (SSHAC), distinguishes between ''aleatory'' and ''epistemic'' uncertainties. The deterministic approach can explicitly recognize epistemic uncertainty just as in the probabilistic approach. The deterministic approach does not explicitly include all components of aleatory variability. The commenter noted that sensitivity analyses are generally intended for addressing epistemic uncertainty, not aleatory variability.

Response: Despite extensive advances in seismic knowledge in recent years by a large and active community of researchers around the world, there are still major gaps in the understanding of the mechanisms that cause earthquakes. These gaps in understanding mean that in any seismic hazard analysis, either deterministic or probabilistic, there are inevitably significant uncertainties in the numerical results. These uncertainties can be classified into two different categories: (1) epistemic uncertainty which is due to lack of knowledge because the scientific understanding is imperfect for the present, but is of a character that in principle is reducible through further research; and (2) aleatory uncertainty which is due to the randomness of seismic events and, in principle, cannot be reduced. As stated in the SSHAC report, ''The division between the two different types of uncertainty, epistemic and aleatory, is somewhat arbitrary, especially at the border between the two. This is because, conceptually, some of the processes and parameters whose uncertainties the NRC will characterize here as aleatory (''random'') may be partially reducible through more elaborate models and/or further study''. As stated further in the SSHAC report, ''the PSHA that does not deal appropriately with both the epistemic and the aleatory uncertainties must be considered inadequate.'' Based on this, the term ''uncertainty'' included in the proposed rule is appropriate.

Revise the Design Earthquake Ground Motion

Comment 17: A commenter stated that performance standards are not clearly articulated in the proposed rule. The commenter also stated that before the design standard is lowered, the performance standards or goals by which the proposed changes were evaluated should first be identified.

Response: The current regulations in § 72.122(b)(2)(i) require that the structures, systems, and components of an ISFSI or MRS must be designed to withstand the effects of natural phenomena, such as earthquakes, without impairing their capability to perform their intended design functions. For earthquakes, these requirements are then supplemented by the §§ 72.102 and 72.103 requirements for the detailed site investigations and consideration of uncertainties in the prediction of earthquakes. This approach is consistent with the Commission's philosophy of using risk-informed, performance-based regulations. In a risk-informed, performance-based approach, the design of the facility is based on considering the risk (potential for adverse consequences) due to an earthquake.

Comment 18: One commenter is concerned that lowering the existing DE may result in a concomitant lowering of the design basis for locally-sourced tsunamis. The commenter is concerned because the most likely scenario for release of radiation in a coastal setting would be damage to an ISFSI or MRS during a major earthquake, followed by inundation of the facility by a tsunami.

Response: Section 72.103(f)(1) requires consideration of actual or potential geologic and seismic effects at the proposed site, including locally-sourced tsunamis. Potential inundation of the facility by a tsunami is required to be addressed in the design of the facility under § 72.122(b)(2). Under the amended rule, the tsunami magnitudes corresponding to the DE would be lower than for a nuclear power plant. However, an earthquake similar in magnitude to the SSE for an NPP would not damage an ISFSI or MRS facility, thus no release of radioactivity would occur even if the facility were inundated by a resulting locally-sourced tsunami.

Comment 19: A commenter stated that in order to issue a coastal development permit in California the State or a local government must make a finding that the proposed ISFSI will minimize risks to life and property in areas of high geologic hazard, and assure stability and structural integrity of the proposed coastal development. The commenter noted that, for the San Onofre Nuclear Generating Station (SONGS) ISFSI, the required finding was able to be made by the State only because the applicant proposed a seismic design standard far in excess of the SSE for the co-located NPP. The commenter indicated that such a finding may not be possible at future ISFSI sites if the applicant submits a design standard lower than those required for an NPP. The commenter stated that the proposed rule change makes approval of coastal development permits in California for future ISFSIs difficult at best.

Response: The NRC sees no reason why the rule would make this finding difficult. The rule ensures adequate protection of public health and safety in all environs. The close proximity of faults or populations are considered in the regulations (for example, the dose requirements contained in §§ 72.104(a) and 72.106(b)). Applying a risk-informed approach to seismic design of ISFSIs takes these factors into account and the analyses indicate that protection of public health and safety are adequately addressed.

Proposed Change to 10 CFR 72.212(b)(2)(i)(B)

Comment 20: Two commenters noted that although the proposed change to 10 CFR 72.212(b)(2)(i)(B) to require that the cask storage pads and areas be designed to adequately support dynamic loads, as well as static loads, of the stored casks, may require more analytical effort than the static load evaluations that some licensees had attempted to utilize in the past, they find the new requirements to be technically correct and support the concept that the seismic evaluation should be conducted using state-of-the-art structural dynamics principles, including consideration of dynamic loads. One commenter had no objection to the portion of the proposed rule that would require design of cask storage pads and areas to adequately account for dynamic loads. Another commenter stated that requiring this evaluation for storage pads and areas clearly improves the assurance of safety.

Response: The commenters support the NRC's decision to require evaluation of dynamic loads for storage cask pads and areas. Further, general licensees currently consider dynamic loads for evaluating the casks, pads and areas to meet the cask design bases in the Certificate of Compliance, as required by 10 CFR 72.212(b)(2)(i)(A); therefore, the rule change will not actually impose a new burden on the general licensees.

Related Regulatory Guide

Comment 21: A commenter stated that Draft Regulatory Guide DG–3021 ''is short on firm standards'' because, although it recommends a DE at a MAPE of 5E–4, it also allows an applicant to demonstrate that the use of a higher probability of exceedance value would not impose any undue radiological risk to public health and safety. Thus, the draft guidance, in the commenter's view, ''leaves open the possibility of an even lower standard for seismic sites.'' Another commenter defends the guidance that an applicant could propose a higher probability of exceedance value as being an exemption to what the commenter sees as the norm being established in DG–3021.

Response: Section 72.103(f)(2)(i) of the rule requires that an applicant include a determination of the DE for the site, considering the results of the investigations required by paragraph (f)(1) and addressing uncertainties through an appropriate analysis, such as a PSHA or suitable sensitivity analyses. Regulatory Guide 3.73 (formerly DG–3021) states that a mean annual probability of exceeding the DE of 5E–4 is recommended to be used in conjunction with the PSHA for determining the DE. As the commenter notes, the draft guidance also indicated that ''[t]he use of a higher reference probability will be reviewed and accepted on a case-by-case basis.'' This statement was made in recognition of the fact that a regulatory guide does not establish legally-binding requirements. An alternative reference probability would not be an exemption from a requirement, but would be an alternative proposal which would need to be demonstrated to be acceptable. Thus, it is conceivable that an applicant could propose a higher MAPE value that the NRC staff would then have to consider. Although this is necessarily the case for recommendations suggested in guidance documents, the NRC did not mean to imply that it viewed an applicant's ability to make the necessary safety case for a higher MAPE as being a likely prospect. To avoid any such implication, that sentence has been removed from the final guidance.

Comment 22: One commenter stated that a DE at a MAPE of 5E–4 (2,000 year return period) is not defensible. The commenter said that there are numerous standards that already use a DE at a MAPE of 4E–4 (2,500 year return period), including DOE Standard 1020–2000. The commenter noted that DOE's standard is inextricably tied to meeting performance and risk goals. Further, the commenter indicated that certain buildings, such as hospitals, must meet a DE at a MAPE of 4E–4 (2,500 year return period), as must interstate bridges in the State of Utah. The commenter stated that, at a minimum, a standard lower than these cannot be adopted.

Response: The NRC disagrees with the commenter that the proposed standard for the DE at a MAPE of 5E–4 (2,000 year return period) is lower than the DOE Standard DOE–STD–1020–2002, or the other standards, such as the International Building Code (IBC–2000 Code).

According to the DOE Standard DOE–STD–1020–2002, ISFSIs can be classified as Performance Category 3 (PC–3) facilities. For PC–3 facilities, the seismic design forces for the DE are initially determined at 90 percent of the DE at a MAPE of 4E–4 (2,500 years return period). This brings the DE levels to approximately a MAPE of 5E–4 (2,000 year return period), specified in the earlier DOE 1020 standard, DOE–STD–1020–94. The Foreword of the DOE–STD–1020–2002 explains the change in the return period as follows:

''It is not the intent of this revision to alter the methodology for evaluating PC–3 facilities, nor to increase the performance goal of PC–3 facilities, by increasing return period for the PC–3 from a 2,000-year earthquake to a 2,500-year earthquake. Rather, the intention is more for convenience to provide a linkage from the NEHRP maps and DOE Standards.''

Therefore, use of the reference probability of 5E–4/yr (2,000 year return period), for the ISFSI or MRS facility DE, would be consistent with that used in the DOE Standard DOE–STD–1020, for similar type facilities.

For the IBC–2000 Code, the commenter is incorrectly comparing the ISFSI or MRS DE at a MAPE of 5E–4 (2,000 year return period), with the Maximum Considered Earthquake (MCE) at a MAPE of 4E–4 (2,500 year return period). The DE, according to the IBC–2000 Code, is two-thirds of the MCE, which is equivalent to a DE at a MAPE of 1.1E–3 (909 year return period) earthquake in the western United States, and a DE at a MAPE of 7E–4 (1,430 year return period) in the eastern United States. Thus, the DE for the ISFSI or MRS facility included in DG–3021 at a MAPE of 5E–4 is greater than the IBC Code DE design level.

The NRC agrees that hospital building structures and bridges having critical national defense functions are designed for the DE at a MAPE of 4E–4 (2,500 year return period). These structures are generally occupied by a significant number of people. Therefore, these structures are designed for loads greater than those for traditional buildings to limit building deformations, and to minimize human losses due to an earthquake. The ISFSI or MRS facility, on the other hand, has a relatively small number of people occupying the Canister Transfer Building at any one time.

Comment 23: A commenter requested that the regulatory guide specify a DE at a MAPE of 1E–4 (10,000 year return period), consistent with the requirement for NPPs. This commenter believes that meeting NPP standards would be easier at an ISFSI or MRS due to the relative simplicity of construction and robust character of the structures as compared to an NPP.

Response: The NRC disagrees with the commenter and believes that the proposed DE at a MAPE of 5E–4 (2,000 year return period) for an ISFSI or MRS facility is adequate for protecting public health and safety. The seismically induced risk from the operation of an ISFSI or MRS is less than from the operation of an NPP, and based on the review of the current seismic design practice, the proposed DE design level is reasonable and consistent with the NRC's policy of risk-informed, performance-based regulations. Details of the NRC's review for the proposed DE level are provided in the report, ''Selection of Design Earthquake Ground Motion Reference Probability''. This report may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr@nrc.gov.*

The NRC agrees with the commenter that the cask structure is simple in construction and robust in character resulting from the design considerations other than earthquake effects. Earthquake loads and the DE level would not govern the cask design. However, this is not the case in the design and stability evaluation of other ISFSI or MRS facility structures, systems, and components, such as the concrete pad, foundation, and the canister transfer building. Designs of these structures, systems, and components depend on the DE level. Further, because of the inherent safety margins in the design criteria in NUREG–1536 and NUREG–1567, the structures, systems, and components designed for a DE at a MAPE of 5E–4 (2,000 year return period) would be able to withstand a DE at a MAPE of 1E–4 (10,000 year return period consistent with the NPP requirements) without impairing the ability to meet the Part 72 dose limits for protecting public health and safety. Therefore, it is an unnecessary burden on the applicant to require the ISFSI or MRS facility to design for a DE at a level consistent with NPP requirements.

Comment 24: Two commenters stated that the seismic design standard (MAPE of 5E–4 (2,000 year return period)) is less protective than the seismic standard for municipal solid waste landfills in California (maximum credible earthquake (MCE) of 4E–4 (2,500 year return period)), and the International Building Code (MCE of 4E–4 (2,500 year return period)), both of which are more stringent than the proposed rule. One commenter is concerned that a DE at a MAPE of 5E–4 (2,000 year return period) may not provide an adequate margin of safety to protect the public.

However, two other commenters stated that the rigor of the seismic evaluation criteria and the conservatism of the seismic design requirements significantly exceed those in modern conventional building codes. One of the commenters stated that the annual probability of unacceptable seismic performance for a dry cask ISFSI designed to a DE at a MAPE of 5E–4 (2,000 year return period) will be substantially less than that of an essential or hazardous facility designed to the modern conventional building code for which the DE was established at 67 percent of the MCE of 4E–4. Another commenter stated that the level of safety for a dry cask storage facility designed to a DE at a MAPE of 5E–4 (2,000 year return period) provides at least twice the level of safety attained by facilities designed under the International Building Code.

Response: The NRC disagrees with the commenters that the seismic design standard (MAPE of 5E–4) is less protective than the seismic standard for municipal solid waste landfills in California (Code of Regulations Section 66264.25(b), and the International Building Code—2000 (IBC–2000). The California standard requires the municipal waste landfills to be designed to withstand the maximum credible earthquake (MAPE of 4E–4) of the IBC–2000 without decreasing the level of public health and environmental protection. The cask and the cask transfer building at an ISFSI or MRS facility, designed to a DE at a MAPE of 5E–4, has the capacity to withstand earthquakes of greater magnitude than the one associated with the MAPE of 4E–4. This is because of the conservatism in the seismic evaluation criteria and of NRC's NUREG–1536 and NUREG–1567, which significantly exceed those in modern conventional building codes. Additionally, the risk of the ISFSI or MRS facility to public health and safety is lower than the risk for hazardous waste and municipal solid waste landfills because the spent nuclear fuel is contained within a sealed steel cask in an isolated facility away from the public, with a controlled boundary at a minimum distance of 100 m. Landfills, on the other hand, may be open and in close proximity to public areas.

Comment 25: Three commenters stated that the proposed rule provided no basis or quantitative analysis to justify lowering the DE to any particular value. One of these commenters indicated that absent any quantitative evidence justifying a particular value, the conservative, precautionary approach of requiring ISFSIs and MRSs to meet the same design standard as a nuclear power plant is most appropriate. One of these commenters noted that the adequacy of the MAPE should be addressed with respect to the change in the DE. The commenter stated that this could be addressed by using the higher proposed MAPE versus what is currently required and then determining if the change in the level of risk of a release is significant or not.

Response: The DE level proposed in the draft regulatory guide was selected based on the fact that the ISFSI or MRS risk is lower than that of an NPP and on the fact that this level is consistent with the hazard levels used in the nuclear industry for similar facilities. Details of the NRC's analyses for establishing the DE level are provided in the report, ''Selection of Design Earthquake Ground Motion Reference Probability''. This report may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr@nrc.gov.*

Comment 26: Two commenters strongly endorsed the proposal to lower the DE. The commenters stated that the DE provided in the draft regulatory guide at a MAPE of 5E–4 (2,000 year return period) provides a level of relief in establishing the DE that is completely consistent with the risk-informed regulation policy and is an excellent example of the application of the policy. One commenter stated that the philosophy of applying a graded approach to seismic design requirements for facilities of differing risks has been in existence for more than 30 years. The commenter described DOE's approach for seismic design requirements for DOE facilities, which span a range of potential risks. The commenter went on to state that based on the amount of radioactive material stored in a large dry cask ISFSI, the resulting classification using the DOE approach would result in a design standard with a MAPE of 5E–4. The commenter stated that considering the minor radiological consequences from a single canister failure and a lack of a credible mechanism to cause such a failure from a seismic event would suggest that this design criteria level is more than adequately conservative for a dry cask ISFSI.

Response: The commenters support the NRC's recommendation of the seismic design earthquake level to a MAPE of 5E–4 (2,000 year return period).

Finding of No Significant Environmental Impact: Availability

Comment 27: Three commenters challenged the assertion that the NRC has considerable experience in licensing dry cask storage systems and analyzing cask behavior. One commenter noted that the NRC has licensed only four ISFSIs in the western U.S., the most seismically active part of the country, and none as close to major plate-boundary faults as the three planned for coastal California. The commenters also said that analytical experience in licensing does not equate with practical experience. One commenter stated that this will only be achieved when an ISFSI experiences strong ground motions as a result of a major earthquake. As a result, the commenter believes that neither the specific nor general licenses issued have been tested.

Response: As discussed in the NRC response to Comment 4, cask stability can be evaluated with adequate reliability by using non-linear dynamic analyses because the concept of free-standing structures is not a new one. One does not need to test all structures prior to using them, provided structures are simple and can be reliably analyzed.

Regulatory Analysis

Comment 28: A commenter noted that the proposed changes impose no new burdens on establishing the DE for an ISFSI over the current requirements in 10 CFR part 72.

Response: The NRC's analysis actually indicates that there would be an overall reduction in the total burden placed on licensees from these changes. The estimate of values and impacts to a specific-license applicant indicates additional costs of $100,000 for addressing uncertainties in seismic hazard analysis. In some cases, ISFSI specific-license applicants have sought exemptions from the design requirements contained in § 72.102, considering site characteristics and other factors. The rule would reduce or eliminate the need for these exemption requests by reducing the DE level for certain structures, systems, and components, resulting in a savings of $150,000 per license applicant. Further, no structures, systems, and components would be required to be designed to withstand a DE at a MAPE of 1E–4 (equivalent to the SSE of an NPP), resulting in lower analytical and certain capital costs. The overall effect of the rule would be a cost savings to new specific-license applicants. However, the amount of these savings is highly site-specific, depending on site characteristics and the specified DE level.

Finally, the rule will change § 72.212(b)(2)(i)(B) to require written evaluations, prior to use, establishing that cask storage pads and areas have been evaluated for the static and dynamic loads of the stored casks. There are no additional costs associated with evaluating cask pads and areas for dynamic loads because general licensees are already required to consider dynamic loads to meet the cask design basis of the Certificate of Compliance under § 72.212(b)(i)(A).

VII. Summary of Final Revisions

This final rule will make the following changes to 10 CFR part 72:

Section 72.9 Information collection requirements: OMB approval

In § 72.9, the list of sections where approved information collection requirements appear is amended to add § 72.103.

Section 72.102 Geological and seismological characteristics (Current Heading)

Section 72.102 Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage (New Heading)

The heading of § 72.102 is revised because § 72.103 is added for ISFSI or MRS applications after the effective date of the rule. Section 72.103 will only apply to dry cask modes of storage. Therefore, the heading of § 72.102 is being modified to show the revised applicability of this section. The requirements of § 72.102 will continue to apply for an ISFSI or MRS using wet modes of storage or dry modes of storage that do not use casks.

The NRC does not intend for existing part 72 licensees to re-evaluate the geological and seismological characteristics for siting and design using the revised criteria in the changes to the regulations. These existing facilities are considered safe because the criteria used in their evaluation have been determined to be safe for NPP licensing, and the seismically induced risk of an ISFSI or MRS is significantly lower than that of an NPP. The change leaves the current § 72.102 in place to preserve the licensing bases of present ISFSIs.

Section 72.103 Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003

The trend towards dry cask storage has resulted in the need for applicants for new licenses to request exemptions from § 72.102(f)(1), which requires that for sites evaluated under the criteria of Appendix A to Part 100, the DE must be equivalent to the SSE for an NPP. By making § 72.102 applicable only to existing ISFSIs and by providing a new § 72.103, the revised rule is intended to preclude the need for exemption requests from new specific-license applicants.

The new requirements in § 72.103 parallel the requirements in § 72.102. However, new specific-license applicants for sites located in either the western U.S. or in the eastern U.S. in areas of known seismic activity, and not co-located with an NPP, for dry cask storage applications, on or after the effective date of this rule, will be required to address the uncertainties in seismic hazard analysis by using a PSHA or sensitivity analyses instead of using the deterministic methods of Appendix A to Part 100 without sensitivity analyses. Applicants located in either the western U.S. or in areas of known seismic activity in the eastern U.S., and co-located with an NPP, have the option of using the PSHA methodology or suitable sensitivity analyses for determining the DE, or using the existing design criteria for the NPP. This change to require an understanding of the uncertainties in the determination of the DE will make the regulations compatible with 10 CFR 100.23 for NPPs and will allow the geological and seismological criteria for ISFSI or MRS dry cask storage facilities to be risk-informed.

New § 72.103(a)(1) provides that sites located in eastern U.S. and not in areas of known seismic activity, will be acceptable if the results from onsite foundation and geological investigation, literature review, and regional geological reconnaissance show no unstable geological characteristics, soil stability problems, or potential for vibratory ground motion at the site in excess of an appropriate response spectrum anchored at 0.2 g. Section 72.103(a)(1) will parallel the requirements currently included in § 72.102(a)(1).

New § 72.103(a)(2) provides that applicants conducting evaluations in accordance with § 72.103(a)(1) may use a standardized DE described by an appropriate response spectrum anchored at 0.25 g. These requirements parallel the requirements currently included in § 72.102(a)(2). Section 72.102(a)(2) provides an alternative to determine a site-specific DE using the criteria and level of investigations required by Appendix A to Part 100. New § 72.103(a)(2) also provides, as an alternative, that a site-specific DE may be determined by using the criteria and level of investigations in new § 72.103(f). Section 72.103(f) is a new provision that requires certain new ISFSI or MRS license applicants to address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, in determining the DE instead of the current deterministic approach in Appendix A to Part 100.

New § 72.103(a)(2) also provides that if an ISFSI or MRS is located at an NPP site, the existing geological and seismological design criteria for the NPP may be used instead of PSHA techniques or suitable sensitivity analyses because the risk due to a seismic event at an ISFSI or MRS is less than that of an NPP. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used to ensure that the seismic design criteria used is based on the latest seismic hazard information at the site.

New § 72.103(b) provides that applicants for licenses for sites located in either the western U.S. or in the eastern U.S. in areas of known seismic activity, must investigate the geological, seismological, and engineering characteristics of the site using the PSHA techniques or suitable sensitivity analyses of new § 72.103(f). If an ISFSI or MRS is located at an NPP site, the existing geological and seismological design criteria for the NPP may be used instead of PSHA techniques or suitable sensitivity analyses because the risk due to a seismic event at an ISFSI or MRS is less than that of an NPP. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used to ensure that the seismic design criteria used is based on the latest seismic hazard information at the site.

New § 72.103(c) is identical to § 72.102(c). Section 72.103(c) requires that sites, other than bedrock sites, must be evaluated for the liquefaction potential or other soil instability due to vibratory ground motion. This is to ensure that an ISFSI or MRS will be adequately supported on a stable foundation during a seismic event.

New § 72.103(d) is identical to § 72.102(d). Section 72.103(d) requires that site specific investigation and laboratory analysis must show that soil conditions are adequate for the proposed foundation loading. This is to ensure that an ISFSI or MRS will be adequately supported on a stable foundation during a seismic event.

New § 72.103(e) is identical to § 72.102(e). Section 72.103(e) requires that in an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred, and that sites with unstable geologic characteristics should be avoided. This is to ensure that sites with minimum deficiencies are selected and that an ISFSI or MRS will be adequately supported on a stable foundation during a seismic event.

New § 72.103(f) describes the steps required for seismic hazard analysis to determine the DE for use in the design of structures, systems, and components of an ISFSI or MRS. The scope of site investigations to determine the geological, seismological, and engineering characteristics of a site and its environs is similar to § 100.23 requirements. Unlike § 72.102(f), which requires the use of the deterministic method of Appendix A to Part 100, new § 72.103(f) requires evaluating uncertainty in seismic hazard analysis by using a probabilistic method, such as the PSHA, or suitable sensitivity analyses, similar to § 100.23 requirements for an NPP.

New § 72.103(f)(1) requires that the geological, seismological, and engineering characteristics of a site and its environs must be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site and to determine the DE. These requirements track existing requirements in § 100.23(c).

New §§ 72.103(f)(2)(i) through (iv) specify criteria for determining the DE for the site, the potential for surface tectonic and nontectonic deformations, the design basis for seismically induced floods and water waves, and other design conditions. In particular, § 72.103(f)(2)(i) provides that a specific-license applicant must address uncertainties in seismic hazard analysis by using appropriate analyses, such as a PSHA or suitable sensitivity analyses, for determining the DE. Sections 72.103(f)(2)(ii) through (iv) track the corresponding requirements in § 100.23(d).

Finally, the new § 72.103(f)(3) provides that regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum. This provision is identical to the requirement currently included in § 72.102(f)(2).

Section 72.212 Conditions of general license issued under § 72.210

Section 72.212(b)(2)(i)(B) is revised to require general licensees to address the dynamic loads of the stored casks in addition to the static loads. The requirements are changed because during a seismic event the cask experiences dynamic inertia loads in addition to the static loads, which are supported by the concrete pad. The dynamic loads depend on the interaction of the casks, the pad, and the foundation. Consideration of the dynamic loads, in addition to the static loads, of the stored casks will ensure that the pad would perform satisfactorily during a seismic event.

The new paragraph also requires consideration of potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion. Depending on the properties of soil and structures, the free-field earthquake acceleration input loads may be amplified at the top of the storage pad. These amplified acceleration input values must be bound by the design bases seismic acceleration values for the cask, specified in the Certificate of Compliance. Liquefaction of the soil and instability during a vibratory motion due to an earthquake may affect the cask stability, and thus must be addressed.

The changes to § 72.212 are intended to require that general licensees perform appropriate load evaluations of cask storage pads and areas to ensure that casks are not placed in an unanalyzed condition. Similar requirements currently exist in § 72.102(c) for an ISFSI specific license and are now in § 72.103(c).

VIII. Criminal Penalties

For the purpose of Section 223 of the Atomic Energy Act (AEA), the Commission is issuing this final rule to amend 10 CFR Part 72 under one or more of sections 161b, 161i, or 161o of the AEA. Willful violations of the rule will be subject to criminal enforcement.

IX. Agreement State Compatibility

Under the ''Policy Statement on Adequacy and Compatibility of Agreement State Programs'' approved by the Commission on June 30, 1997, and published in the **Federal Register** on September 3, 1997 (62 FR 46517), this rule is classified as Compatibility Category ''NRC.'' Compatibility is not required for Category ''NRC'' regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the AEA of 1954, as amended (AEA), or the provisions of Title 10 of the Code of Federal Regulations. Although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with the particular State's administrative procedure laws, but does not confer regulatory authority on the State.

X. Voluntary Consensus Standards

The National Technology Transfer Act of 1995 (Pub. L. 104–113) requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC is presenting amendments to its regulations in 10 CFR part 72 for the geological and seismological criteria of a dry cask independent spent fuel storage facility to make them commensurate with the risk of the facility. This action does not constitute the establishment of a standard that establishes generally applicable requirements.

XI. Finding of No Significant Environmental Impact: Availability

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR part 51, that this rule is not a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required.

The Commission concluded, based on an environmental assessment, that no significant environmental impact would result from this rulemaking. In comparison with an NPP, an operating ISFSI or MRS is a passive facility in which the primary activities are waste receipt, handling, and storage. An ISFSI or MRS does not have the variety and complexity of active systems necessary to support an operating NPP. After the spent fuel is in place, an ISFSI or MRS is essentially a static operation and, during normal operations, the conditions required for the release and dispersal of significant quantities of radioactive materials are not present. There are no high temperatures or pressures present during normal operations or under design basis accident conditions to cause the release and dispersal of radioactive materials. This is primarily due to the low heat generation rate of spent fuel after it has decayed for more than one year before storage in an ISFSI or MRS and the low inventory of volatile radioactive materials readily available for release to the environs. The long-lived nuclides present in spent fuel are tightly bound in the fuel materials and are not readily dispersible. The short-lived volatile nuclides, such as I–131, are no longer present in aged spent fuel stored at an ISFSI or MRS. Furthermore, even if the short-lived nuclides were present during an event of a fuel assembly rupture, the canister surrounding the fuel assemblies would confine these nuclides.

The standards in part 72 Subparts E ''Siting Evaluation Factors,'' and F ''General Design Criteria,'' ensure that the dry cask storage designs are very rugged and robust. The casks must maintain structural integrity during a variety of postulated non-seismic events, including cask drops, tip-over, and wind driven missile impacts. These non-seismic events challenge cask integrity significantly more than seismic events. Therefore, the casks have substantial design margins to withstand forces from a seismic event greater than the design earthquake.

Hence, the seismically induced radiological risk associated with an ISFSI or MRS is less than the risk associated with an NPP.

The determination of the environmental assessment is that there will be no significant environmental impact due to the rule changes because the same level of safety would be maintained by the new requirements, taking into account the lesser risk from an ISFSI or MRS. The NRC requested public comments on the environmental assessment for this rule.

XII. Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq*). These requirements were approved by the Office of Management and Budget, approval number 3150–0132.

Because the rule will reduce existing information collection requirements, the public burden for these information collections is expected to be decreased by 55 hours per licensee. This reduction includes the time required for reviewing instructions, searching existing data sources, gathering and maintaining the data needed and completing and reviewing the information collection. Send comments on any aspect of these information collections, including suggestions for further reducing the burden, to the Records Management Branch (T–6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by Internet electronic mail at *infocollects@nrc.gov*; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202, (3150–0132), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis

The Commission has prepared a Regulatory Analysis (RA) entitled: ''Regulatory Analysis of Geological and Seismological Characteristics for Design of Dry Cask Independent Spent Fuel Storage Installations.'' The RA examines the costs and benefits of the alternatives considered by the Commission. The RA may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC's PDR reference staff at 1–800–397–4209, 301–415–4737, or by email to *pdr@nrc.gov*.

XIV. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule does not have a significant economic impact on a substantial number of small entities. This rule affects applicants for a Part 72 specific license, and general licensees on or after the effective date of the rule for an ISFSI or MRS. These companies do not generally fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121.

XV. Backfit Analysis

The NRC has determined that the backfit rule, 72.62, does not apply to the changes in §§ 72.9, 72.102, and 72.103 because they do not involve any provisions that would impose backfits as defined in the backfit rule. Therefore, a backfit analysis is not required for these provisions.

Section 72.212(b)(2)(i)(B) currently requires evaluations of static loads of the stored casks for design of the cask storage pads and areas (foundation). The revision to this section will require general licensees also to address the dynamic loads of the stored casks. During a seismic event, the cask storage pads and areas experience dynamic loads in addition to static loads. The dynamic loads depend on the interaction of the casks, cask storage pads, and areas. Consideration of the dynamic loads of the stored casks, in addition to the static loads, for the design of the cask storage pads and areas will ensure that the cask storage pads and areas will perform satisfactorily in the event of an earthquake.

The revision will also require consideration of potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion. Depending on the properties of soil and structures, the free-field earthquake acceleration input loads may be amplified at the top of the storage pad. These amplified acceleration input values must be bound by the design bases seismic acceleration values for the cask specified in the Certificate of Compliance. The soil liquefaction and instability during a vibratory motion due to an earthquake may affect the cask stability.

The changes to § 72.212(b)(2)(i)(B) will impact procedures required to operate an ISFSI and, therefore, implicate the backfit rule. The changes will require that general licensees perform appropriate analyses to assure that the cask seismic design bases bound the specific site seismic conditions, and that casks are not placed in an unanalyzed condition. Therefore, these changes are necessary to assure adequate protection to occupational or public health and safety. Although the Commission is imposing this backfit because it is necessary to assure adequate protection to occupational or public health and safety, the changes to § 72.212 will not actually impose new burden on the general licensees because they currently need to consider dynamic loads to meet the requirements in § 72.212(b)(2)(i)(A). Section 72.212(b)(2)(i)(A) requires general licensees to perform written evaluations to meet conditions set forth in the cask Certificate of Compliance. These Certificates of Compliance require that dynamic loads, such as seismic and tornado loads, be evaluated to meet the cask design bases. Because the general licensees currently evaluate dynamic loads for evaluating the casks, pads and areas, the changes to § 72.212(b)(2)(i)(B) will not actually require any general licensees presently operating an ISFSI to re-perform any written evaluations previously undertaken.

XVI. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

List of Subjects In 10 CFR Part 72

Administrative practice and procedure, Criminal penalties, Manpower training programs, Nuclear materials, Occupational safety and health, Penalties, Radiation protection, Reporting and recordkeeping requirements, Security measures, Spent fuel, Whistleblowing.

■ For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; and 5 U.S.C. 552 and 553; the

NRC is adopting the following amendments to 10 CFR part 72.

PART 72—LICENSING REQUIREMENTS FOR THE INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE, AND REACTOR-RELATED GREATER THAN CLASS C WASTE

■ 1. The authority citation for Part 72 continues to read as follows:

Authority: Secs. 51, 53, 57, 62, 63, 65, 69, 81, 161, 182, 183, 184, 186, 187, 189, 68 Stat. 929, 930, 932, 933, 934, 935, 948, 953, 954, 955, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2071, 2073, 2077, 2092, 2093, 2095, 2099, 2111, 2201, 2232, 2233, 2234, 2236, 2237, 2238, 2282); sec. 274, Pub. L. 86–373, 73 Stat. 688, as amended (42 U.S.C. 2021); sec. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); Pub. L. 95–601, sec. 10, 92 Stat. 2951 as amended by Pub. L. 102–486, sec. 7902, 106 Stat. 3123 (42 U.S.C. 5851); sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332); secs. 131, 132, 133, 135, 137, 141, Pub. L. 97–425, 96 Stat. 2229, 2230, 2232, 2241, sec. 148, Pub. L. 100–203, 101 Stat. 1330–235 (42 U.S.C. 10151, 10152, 10153, 10155, 10157, 10161, 10168).

Section 72.44(g) also issued under secs. 142(b) and 148(c), (d), Pub. L. 100–203, 101 Stat. 1330–232, 1330–236 (42 U.S.C. 10162(b), 10168(c), (d)). Section 72.46 also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97–425, 96 Stat. 2230 (42 U.S.C. 10154). Section 72.96(d) also issued under sec. 145(g), Pub. L. 100–203, 101 Stat. 1330–235 (42 U.S.C. 10165(g)). Subpart J also issued under secs. 2(2), 2(15), 2(19), 117(a), 141(h), Pub. L. 97–425, 96 Stat. 2202, 2203, 2204, 2222, 2224, (42 U.S.C. 10101, 10137(a), 10161(h)). Subparts K and L are also issued under sec. 133, 98 Stat. 2230 (42 U.S.C. 10153) and sec. 218(a), 96 Stat. 2252 (42 U.S.C. 10198).

■ 2. In § 72.9, paragraph (b) is revised to read as follows:

§ 72.9 Information collection requirements: OMB approval.

* * * * *

(b) The approved information collection requirements contained in this part appear in §§ 72.7, 72.11, 72.16, 72.22 through 72.34, 72.42, 72.44, 72.48 through 72.56, 72.62, 72.70, through 72.82, 72.90, 72.92, 72.94, 72.98, 72.100, 72.102, 72.103, 72.104, 72.108, 72.120, 72.126, 72.140 through 72.176, 72.180 through 72.186, 72.192, 72.206, 72.212, 72.216, 72.218, 72.230, 72.232, 72.234, 72.236, 72.240, 72.242, 72.244, 72.248.

■ 3. The heading of § 72.102 is revised to read as follows:

§ 72.102 Geological and seismological characteristics for applications before October 16, 2003 and applications for other than dry cask modes of storage.

* * * * *

■ 4. A new § 72.103 is added to read as follows:

§ 72.103 Geological and seismological characteristics for applications for dry cask modes of storage on or after October 16, 2003.

(a)(1) East of the Rocky Mountain Front (east of approximately 104° west longitude), except in areas of known seismic activity including but not limited to the regions around New Madrid, MO; Charleston, SC; and Attica, NY; sites will be acceptable if the results from onsite foundation and geological investigation, literature review, and regional geological reconnaissance show no unstable geological characteristics, soil stability problems, or potential for vibratory ground motion at the site in excess of an appropriate response spectrum anchored at 0.2 g.

(2) For those sites that have been evaluated under paragraph (a)(1) of this section that are east of the Rocky Mountain Front, and that are not in areas of known seismic activity, a standardized design earthquake ground motion (DE) described by an appropriate response spectrum anchored at 0.25 g may be used. Alternatively, a site-specific DE may be determined by using the criteria and level of investigations required by paragraph (f) of this section. For a site with a co-located nuclear power plant (NPP), the existing geological and seismological design criteria for the NPP may be used. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used.

(b) West of the Rocky Mountain Front (west of approximately 104° west longitude), and in other areas of known potential seismic activity east of the Rocky Mountain Front, seismicity must be evaluated by the techniques presented in paragraph (f) of this section. If an ISFSI or MRS is located on an NPP site, the existing geological and seismological design criteria for the NPP may be used. If the existing design criteria for the NPP is used and the site has multiple NPPs, then the criteria for the most recent NPP must be used.

(c) Sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.

(d) Site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.

(e) In an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.

(f) Except as provided in paragraphs (a)(2) and (b) of this section, the DE for use in the design of structures, systems, and components must be determined as follows:

(1) *Geological, seismological, and engineering characteristics.* The geological, seismological, and engineering characteristics of a site and its environs must be investigated in sufficient scope and detail to permit an adequate evaluation of the proposed site, to provide sufficient information to support evaluations performed to arrive at estimates of the DE, and to permit adequate engineering solutions to actual or potential geologic and seismic effects at the proposed site. The size of the region to be investigated and the type of data pertinent to the investigations must be determined based on the nature of the region surrounding the proposed site. Data on the vibratory ground motion, tectonic surface deformation, nontectonic deformation, earthquake recurrence rates, fault geometry and slip rates, site foundation material, and seismically induced floods and water waves must be obtained by reviewing pertinent literature and carrying out field investigations. However, each applicant shall investigate all geologic and seismic factors (for example, volcanic activity) that may affect the design and operation of the proposed ISFSI or MRS facility irrespective of whether these factors are explicitly included in this section.

(2) *Geologic and seismic siting factors.* The geologic and seismic siting factors considered for design must include a determination of the DE for the site, the potential for surface tectonic and nontectonic deformations, the design bases for seismically induced floods and water waves, and other design conditions as stated in paragraph (f)(2)(iv) of this section.

(i) Determination of the Design Earthquake Ground Motion (DE). The DE for the site is characterized by both horizontal and vertical free-field ground motion response spectra at the free ground surface. In view of the limited data available on vibratory ground motions for strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The DE for the site is determined considering the results of the investigations required by paragraph (f)(1) of this section. Uncertainties are inherent in these estimates and must be addressed through an appropriate analysis, such as a probabilistic seismic hazard analysis (PSHA) or suitable sensitivity analyses.

(ii) Determination of the potential for surface tectonic and nontectonic deformations. Sufficient geological, seismological, and geophysical data must be provided to clearly establish if there is a potential for surface deformation.

(iii) Determination of design bases for seismically induced floods and water waves. The size of seismically induced floods and water waves that could affect a site from either locally or distantly generated seismic activity must be determined.

(iv) Determination of siting factors for other design conditions. Siting factors for other design conditions that must be evaluated include soil and rock stability, liquefaction potential, and natural and artificial slope stability. Each applicant shall evaluate all siting factors and potential causes of failure, such as, the physical properties of the materials underlying the site, ground disruption, and the effects of vibratory ground motion that may affect the design and operation of the proposed ISFSI or MRS.

(3) Regardless of the results of the investigations anywhere in the continental U.S., the DE must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

■ 5. In § 72.212, paragraph (b)(2)(i)(B) is revised to read as follows:

§ 72.212 Conditions of general license issued under § 72.210.

* * * * *

(b) * * *

(2) * * *

(i) * * *

(B) Cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion; and

* * * * *

Dated at Rockville, Maryland, this 10th day of September, 2003.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,

Secretary for the Commission.

[FR Doc. 03–23553 Filed 9–15–03; 8:45 am]

BILLING CODE 7590–01–P

ENVIRONMENTAL PROTECTION AGENCY

40 CFR Part 52

[WI111–1a; FRL–7547–5]

Approval and Promulgation of Implementation Plans; Wisconsin

AGENCY: Environmental Protection Agency (EPA).

ACTION: Direct final rule.

SUMMARY: The Environmental Protection Agency is approving a revision to the Wisconsin particulate matter (PM) State Implementation Plan (SIP) submitted by the Wisconsin Department of Natural Resources (WDNR) on October 7, 2002. The request is approvable because it satisfies the requirements of the Clean Air Act (Act). The rationale for the approval and other information are provided in this document.

DATES: This rule is effective on November 17, 2003, unless EPA receives adverse written comments by October 16, 2003. If EPA receives adverse comments, EPA will publish a timely withdrawal of the rule in the **Federal Register** and inform the public that the rule will not take effect.

ADDRESSES: You may inspect copies of the documents relevant to this action during normal business hours at the following location: Regulation Development Section, Air Programs Branch, (AR–18J), U.S. Environmental Protection Agency, Region 5, 77 West Jackson Boulevard, Chicago, Illinois 60604. Please contact Christos Panos at (312) 353–8328 before visiting the Region 5 office.

Send written comments to: Carlton Nash, Chief, Regulation Development Section, Air Programs Branch, (AR–18J), U.S. Environmental Protection Agency, Region 5, 77 West Jackson Boulevard, Chicago, Illinois 60604.

Comments may also be submitted electronically or through hand delivery/courier, please follow the detailed instructions described in Part (I)(B)(1)(i) through (iii)of the **SUPPLEMENTARY INFORMATION** section.

FOR FURTHER INFORMATION CONTACT: Christos Panos, Regulation Development Section, Air Programs Branch (AR–18J), United States Environmental Protection Agency, Region 5, 77 West Jackson Boulevard, Chicago, Illinois 60604, (312) 353–8328. *panos.christos@epa.gov.*

SUPPLEMENTARY INFORMATION: This Supplementary Information section is organized as follows:

I. General Information

A. How Can I Get Copies of This Document and Other Related Information?

1. The Regional Office has established an official public rulemaking file available for inspection at the Regional Office. EPA has established an official public rulemaking file for this action under ''Region 5 Air Docket WI111''. The official public file consists of the documents specifically referenced in this action, any public comments received, and other information related to this action. Although a part of the official docket, the public rulemaking file does not include Confidential Business Information (CBI) or other information whose disclosure is restricted by statute. The official public rulemaking file is the collection of materials that is available for public viewing at the Air Programs Branch, Air and Radiation Division, EPA Region 5, 77 West Jackson Boulevard, Chicago, Illinois 60604. EPA requests that if at all possible, you contact the contact listed in the **FOR FURTHER INFORMATION CONTACT** section to schedule your inspection. The Regional Office's official hours of business are Monday through Friday, 8:30 to 4:30 excluding Federal holidays.

2. *Electronic Access.* You may access this **Federal Register** document electronically through the Regulations.gov Web site located at *http://www.regulations.gov* where you can find, review, and submit comments on Federal rules that have been published in the **Federal Register**, the Government's legal newspaper, and are open for comment.

For public commenters, it is important to note that EPA's policy is that public comments, whether submitted electronically or in paper, will be made available for public viewing at the EPA Regional Office, as EPA receives them and without change, unless the comment contains copyrighted material, CBI, or other information whose disclosure is restricted by statute. When EPA identifies a comment containing copyrighted material, EPA will provide